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P. J. Mellinger
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August 1984

Prepared for the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830

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1.0 SUMMARY

In order to investigate the impact of the Environmental Protection Agency's regulations in the Final Environmental Statement - 40 CFR 190 - Environmental Protection Requirements for Normal Operation of Activities in the Uranium Fuel Cycle (EPA 1976b), a health risk assessment was done at Pacific Northwest Laboratory for the Department of Energy to investigate the comparative risks involved in the routine release of ^{85}Kr from nuclear fuel reprocessing operations to the environment were compared to those resulting from the capture and storage of ^{85}Kr.

Krypton-85 is one of the fission products in fuel elements that are normally released when fuel is reprocessed. Instead of releasing the ^{85}Kr to the environment, it can be captured, immobilized and stored. Two alternative methods of capturing ^{85}Kr, cryogenic distillation and fluorocarbon absorption, and one method of immobilizing the captured gas, ion implantation/sputtering, were theoretically incorporated into a representative fuel reprocessing plant, the Barnwell Nuclear Fuel Plant (BNFP), for this study, even though there are no known plans to start up this facility.

For this study we treat the occupational workforce the same as general public and therefore compare their radiation doses accordingly. Workforce and population radiation doses were calculated and analyzed for operating and maintaining the two capture options for immobilization. Those doses were then compared to the public doses from routine release of ^{85}Kr.

The average occupationally exposed worker was estimated to receive about 400 to 600 mrem/yr (3 to 4 man-rem/yr) from ^{85}Kr recovery and immobilization activities. This dose is a factor of 20,000 to 30,000 higher than the estimated dose to the maximum offsite individual (0.02 mrem/yr), and a factor of 130,000 to 200,000 higher than the dose received by the average member of the 50-mile population (0.003 mrem/yr or 2.4 man-rem/yr) from routine release of all ^{85}Kr.

A National Academy of Sciences report (NAS 1980), referred to here as BEIR III, discussed several models for estimating lifetime health risks. We chose to use the linear-quadratic relative risk model because we believe it is the most realistic. Unfortunately, no adequate health risk data exist for comparing health effects for exposure rates of approximately 500 mrem/year and exposure rates of approximately 0.003 mrem/year, the situations of interest here.

Given the uncertainties in the models used to generate lifetime risk numbers (0.02-0.027 radiation induced fatal cancers expected in the occupational workforce and 0.017 fatal cancers in the general population), the differences in total risks for the three situations, (i.e., no-capture and two-capture alternatives) cannot be considered meaningful. It is possible that no risks would occur from any of the three situations. There is certainly no reason to conclude that risks from ^{85}Kr routinely released to the environment are greater than those that would result from the other two situations considered.

Present regulations mandate recovery and disposal of ^{85}Kr from the off gases of a facility reprocessing spent fuel from commercial sources. Because of the lack of a clear-cut indication that recovery would be beneficial, it does not seem prudent to burden the facilities with a requirement for ^{85}Kr recovery, at least until operating experience demonstrates the incentive. The probable high aging of the early fuel to be processed and the higher dose resulting from the release of the unregulated ^{3}H and ^{14}C (Exxon Nuclear Co. 1977) also encourage delaying implementation of the ^{85}Kr recovery in the early plants.

This document provides useful data that will be used in a companion document that reviews EPA calculational methodology embodied within 40 CFR 190 requiring ^{85}Kr recovery for FRPs.

2.0 INTRODUCTION

In the early 1970's, rapid growth of the nuclear power industry was predicted. The Environmental Protection Agency (EPA) projected that there would be 1000 GWe of installed nuclear capacity by the year 2000 (EPA 1973). In order to minimize public exposure to radiation from the operation of the nuclear fuel cycle, the EPA issued a "Draft Environmental Statement for a Proposed Rulemaking Action concerning the Environmental Radiation Protection Requirements for Normal Operations of Activities in the Uranium Fuel Cycle," (EPA 1975). This document was followed on January 5, 1976, by "Environmental Radiation Protection for Nuclear Power Operations - Proposed Standards (40 CFR 190) Supplementary Information" (EPA 1976a). These documents received extensive external review, and numerous substantive comments were submitted to the Agency. On November 1, 1976, the EPA issued the Final Environmental Statement - 40 CFR 190 - Environmental Radiation Protection Requirements for Normal Operations of Activities in the Uranium Fuel Cycle in two volumes (EPA 1976b). The provisions of 40 CFR 190 apply to radiation doses received by members of the general public and to releases of selected radionuclides that result from operations of the nuclear fuel cycle.

One of the radionuclides specified in 40 CFR 190 is ^{85}Kr, which is a fission product in fuel elements irradiated in nuclear power plants. Because almost all of the ^{85}Kr remains in spent fuel elements until it decays or until the fuel elements are reprocessed, nearly all the ^{85}Kr released from the nuclear fuel cycle has come from fuel reprocessing plants (FRPs).

The purpose of FRPs is to chemically separate and recover uranium and plutonium from spent commercial power reactor fuel. In the first stage of reprocessing, certain components in the spent fuel are dissolved in hot nitric acid. During this acid dissolution stage, ^{85}Kr is released to the dissolver off-gas (DOG) system. Historically, the krypton contained in the DOG has been routinely released to the environment from fuel reprocessing operations.

The EPA's regulations limit radiation doses to any member of the public to 25 mrem/yr to the whole body and to any organ except the thyroid gland, which is limited to 75 mrem/yr, from normal routine releases from a fuel cycle site (EPA 1976b). These regulations also limit the ^{85}Kr release to 50,000 Ci/GWe-yr of electrical energy produced from fuel irradiated after January 1, 1983. In order to comply with these regulations, alternatives to normal release of ^{85}Kr have been developed. Instead of releasing the DOG to the environment, the radionuclide-containing gas can be treated where the ^{85}Kr is captured, concentrated, immobilized, and stored for long periods of time. A health risk assessment was therefore done at Pacific Northwest Laboratory (PNL) for the U.S. Department of Energy (DOE) in order to compare the potential risks to the public from routine release of ^{85}Kr with the risks to the workforce from capture and immobilization of ^{85}Kr at a reference FRP. We proceeded with the assumption that at microrem/yr doses individual population segment radiosensitivities are lost in the uncertainties of the analysis. Therefore, individuals in the workforce can be compared to members of the general public because, as humans, they have essentially the same risks of dying of cancer from radiation exposure.

In the early 1970's when 40 CFR 190 was issued, the existing Nuclear Fuel Services FRP, located at West Valley, New York, had shut down for a planned facility expansion. The Barnwell Nuclear Fuel Plant, Barnwell, South Carolina, and the Midwest Fuel Recovery Plant, Morris, Illinois were under construction, and Exxon Nuclear had announced plans to construct a 2000-t/yr plant in eastern Tennessee. The predicted growth in the nuclear industry has not occurred, and, in fact, projections are still being revised downward; in the three years from 1980 to 1982 (DOE 1981, 1982, 1983), projections for the year 2000 have dropped from 180 GWe (in 1980) to 165 GWe (in 1981) to 130 GWe (in 1982). Now it is likely that instead of the predicted 18 FRPs operating by the year 2000 (EPA 1973), there could be none. Because there are no FRPs in operation, spent fuel is being stored instead of reprocessed; the longer spent fuel is stored before it is reprocessed, the less ^{85}Kr would be released during reprocessing because of natural radioactive decay. The need to capture and store ^{85}Kr should be reviewed in regard to the actual number of operating FRPs that will contribute to exposing the public to radiation from the nuclear fuel cycle.

For this study, the Barnwell Nuclear Fuel Plant (BNFP) was chosen as a representative FRP because it most closely resembles an operating FRP. The BNFP has been constructed and cold tested but has never been operated. Two alternative methods of capturing ^{85}Kr (cryogenic distillation and fluorocarbon absorption) and one method of immobilizing the captured radionuclide (ion implantation/sputtering) were theoretically incorporated into the representative BNFP. The increased radiation dose commitment to the operation and maintenance workforce at the reference FRP due to capturing and immobilizing the ^{85}Kr was compared to the public radiation dose commitment that would result from the routine release of all the ^{85}Kr to the atmosphere. This document provides useful data that will be used in a companion document that reviews EPA calculational methodology embodied within 40 CFR 190.

3.0 THE REPRESENTATIVE NUCLEAR FUEL REPROCESSING PLANT

In 1968, Allied-General Nuclear Services (AGNS) was formed by Allied Chemical and Gulf Oil to construct and operate the Barnwell Nuclear Fuel Plant (BNFP) to chemically process spent nuclear fuel. The BNFP construction permit was granted by the Atomic Energy Commission (AEC) in 1970 and construction began in 1971. The BNFP is located on the eastern edge of the Department of Energy's (DOE) Savannah River Plant and seven and one-half miles west of the town of Barnwell, South Carolina (Figure 3.1). Major facilities of the BNFP are shown in Figure 3.2. Additional detail can be found in the BNFP Environmental Report (AGNS 1971).

The purpose of the BNFP is to chemically separate and recover uranium and plutonium from spent commercial power reactor fuel. The plant was scheduled for startup in early 1974 at a nominal 5-t/day (1500 tU/yr) throughput. The BNFP incorporated a chop-leach head-end, and used a 3-cycle PUREX solvent extraction system with pulsed column and centrifugal contactors. Both contact and remote maintenance were designed into this plant. The BNFP has been cold tested using uranium, but has not processed any radioactive spent fuel. During cold testing, some desirable minor modifications were identified and some modifications were made. For various reasons, the owner/operator has elected to terminate further activities at the facility. The technical staff departed in August of 1983 and a reduced maintenance staff stayed on until November 1983.

Historically, the krypton contained in the dissolver off gas has been routinely released to the environment from fuel reprocessing operations. There is an alternative to routine release of all the ^{85}Kr and that is to capture, concentrate, and store the gas for long periods of time. Several sets of alternative methods to recover and store the radioactive gas were considered by Mellinger et al. (1980) for a generic FRP. This effort is being refined here by theoretically placing a cryogenic distillation or a fluorocarbon absorption system and an ion implantation/sputtering facility at the BNFP site. A defined population distribution and meteorological data are used from the BNFP site environment (Chapter 4).

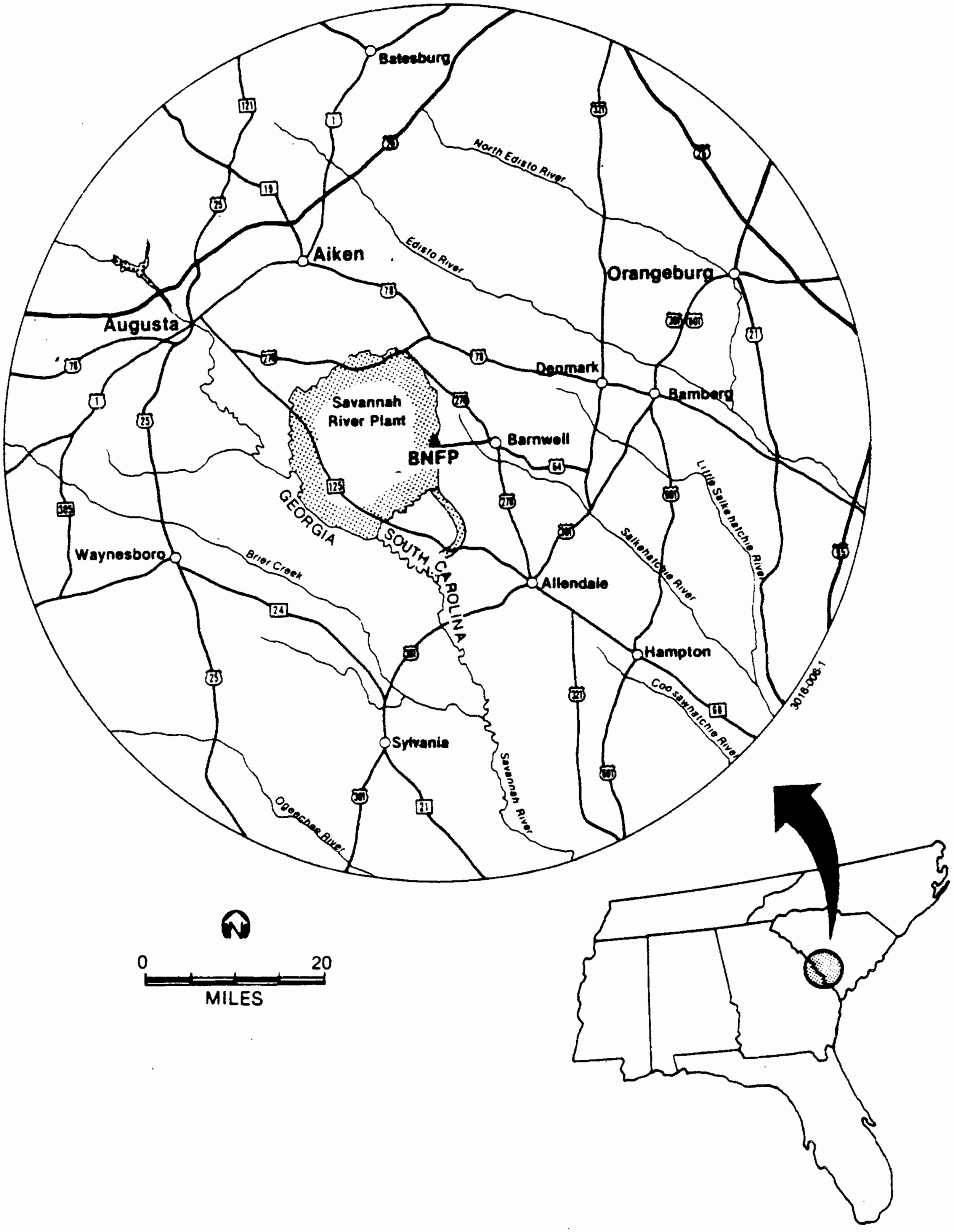


FIGURE 3.1. BNFP Site Location

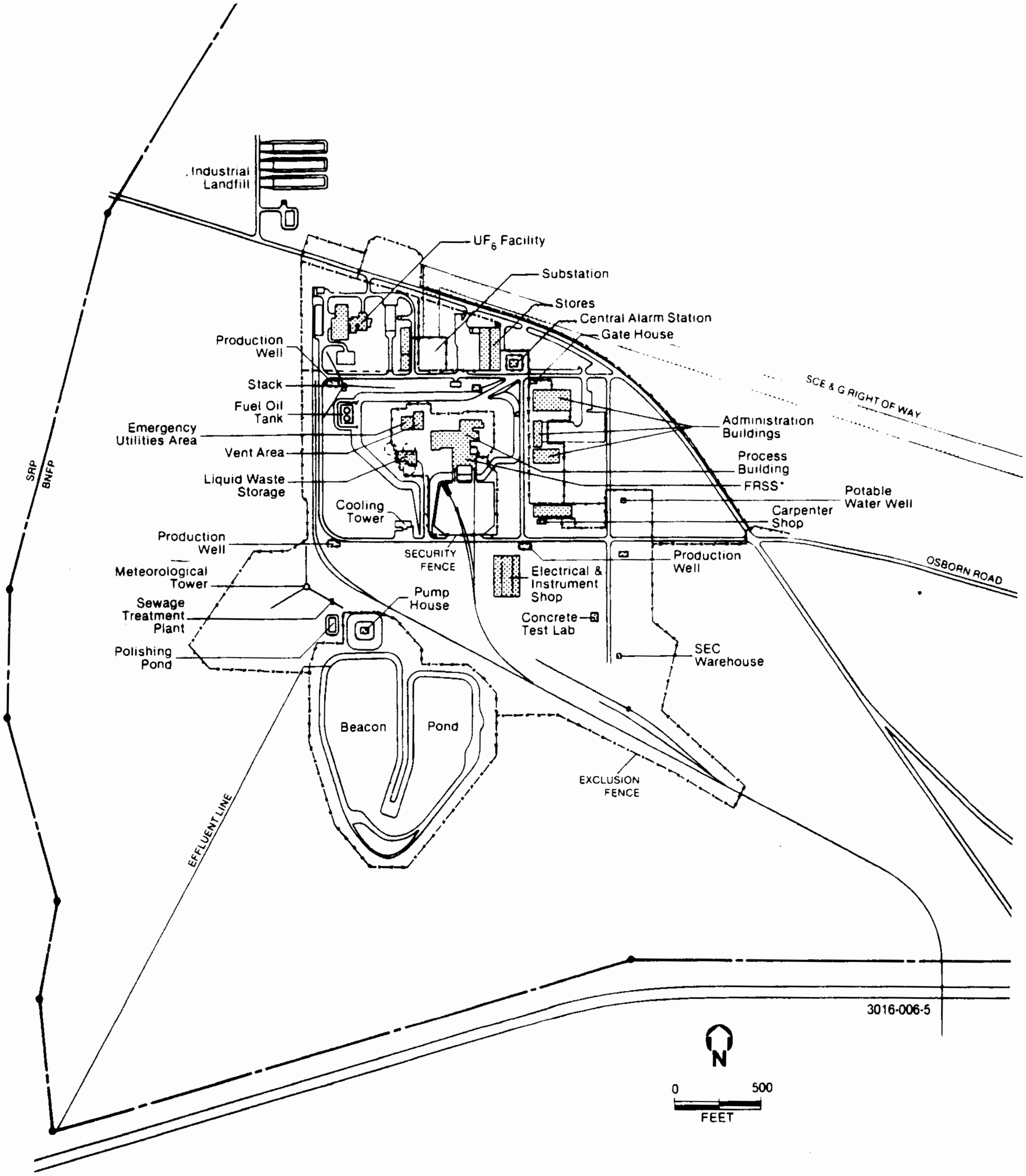


FIGURE 3.2. Plot Plan of the Barnwell Plant (AGNS 1981)

This health risk assessment is being carried out as if the BNFP were in full operation and employed a noble gas recovery, immobilization, and storage system. The following are brief descriptions of the PUREX process, whose head-end and dissolver release the ^{85}Kr from the fuel; the cryogenic distillation system and the alternative fluorocarbon absorption system that concentrate the ^{85}Kr; the ion implantation/sputtering process that immobilizes the captured gas; and the storage scenario.

3.1 PUREX PROCESS

The BNFP is designed to use a solvent extraction process to recover 1500 t/yr of uranium. A chop-leach head-end prepares fuel for dissolution in hot nitric acid. The uranium and plutonium in the solution are then separated and purified in three solvent extraction cycles. The separation process currently used commercially is the aqueous liquid-liquid solvent extraction (PUREX) process developed for military purposes. The PUREX process converts the materials of interest into aqueous solutions by acid dissolution (primarily HNO_3). During acid dissolution, the ^{85}Kr is released to the dissolver off-gas system (DOG). The first cycle separates uranium and plutonium from fission products (generally 30% tributyl phosphate (TBP) in a kerosene-type diluent). The heart of the PUREX process consists of two separate solvent extraction cycles, where the separated uranium and plutonium streams are decontaminated and further purified and concentrated.

3.2 KRYPTON-85 RELEASES

Krypton-85 is formed as a fission product in fuel elements of nuclear power plants. The vast majority of the krypton is retained in the fuel element and would only be released if the fuel was reprocessed. Based on calculations using the computer code ORIGEN (Bell 1973), the quantities of fission products present in recently discharged spent nuclear fuel have been previously reported (ORNL 1970; Exxon Nuclear 1977; and Brown, Knecht and Thomas 1978). For ^{85}Kr, this value ranges from 8,500 to 11,200 Ci/tHM (curies/metric ton of heavy metal) in the fuel. For fuel irradiated at a specific power of 30 MW/tHM with a total burnup of 33,000 MWd/tHM, the calculated ^{85}Kr content 160 days after discharge of the fuel from the reactor is about 10,500 Ci/tHM. A small percentage of fuel elements develop minute leaks in the fuel cladding during irradiation and handling, releasing <1% of the noble gases to the reactor off-gas system. Therefore, the routine releases of ^{85}Kr from commercial LWRs are insignificant when compared to the potential releases from fuel reprocessing operations. It is shown in Chapter 5 that FRP releases can also be insignificant.

As of 1983, the U.S. had accumulated about 9000 t of spent nuclear fuel ranging in age from "fresh" to more than 12 years old. It is unrealistic to assume that, when the U.S. designs, constructs, and licenses a commercial fuel reprocessing plant, "fresh" fuel will be reprocessed when such a tremendous backlog of aged fuel exists. A sensitivity analysis on ^{85}Kr content of aged fuel under three reprocessing scenarios is presented in Appendix A. The decay algorithm (used in Appendix A) requires as input a decay time that is determined by subtracting the year that the spent fuel was produced from the year that the spent fuel is expected to be reprocessed. The decay algorithm uses only one decay time for each year of reprocessing and not for each year the fuel was produced, even though several years worth of fuel production may be reprocessed in the same year. If more than one year's production of spent fuel is reprocessed in one year, then the decay time of the newest spent fuel is used by the decay algorithm. This is conservative, since some of the fuel reprocessed under this scenario has a longer decay time and, therefore, less radioactivity. Under this scenario, the decay algorithm actually over-predicts the ^{85}Kr inventory.

The earliest realistic date that a commercial FRP could become operational would be about the year 2000. At that time, if the oldest fuel is reprocessed first, the facility will never reprocess fuel aged less than about 30 years. Therefore, the routine annual unperturbed release for a 1500-t/yr FRP reprocessing aged fuel is conservatively estimated to be about 3×10^6 Ci ^{85}Kr. This is different than the EPA basis for 40 CFR 190 which assumed 160-day-cooled fuel or about 17×10^6 Ci ^{85}Kr.

3.3 CRYOGENIC DISTILLATION

Cryogenic distillation provides an effective, continuous, small-size system for separating krypton and xenon from DOGs by utilizing different boiling points of the main components. It is expected to exhibit a decontamination factor (dF) of at least 1000 (EPA 1976c). However, overall efficiency for removal of krypton from the plant is expected to be lower because of potential leakage through the system during maintenance, startup, and shutdown operations. Therefore, an effective plant dF of between 10 and 100 (release fraction of 0.1 to 0.01) has been projected for routine operation of a cryogenic distillation system (EPA 1977). One vendor warrants its product for a krypton dF of 10,000 (Airco 1976). A demonstration system on the scale of a commercial facility is required to establish the practical limits for ^{85}Kr recovery from commercial fuel reprocessing facilities.

Cryogenic distillation has been done for research and industrial application at the Idaho Chemical Processing Plant (ICPP) since 1958 (Bendixsen and Offutt 1969). Although the ICPP unit was not designed to achieve the high ^{85}Kr recovery efficiencies necessary for environmental control under current law (EPA 1976b), its operation has indicated such potential.

Additional experience will be gained from the recent addition of a cryogenic distillation system to a foreign pilot scale FRP. An Airco Cryoplants, Inc., cryogenic distillation process design was incorporated by Kobe Steel into the Tokai-Mura pilot scale reprocessing plant in Japan. The system completed cold testing in 1983, but has not yet undergone hot testing.

3.3.1 Process and Facilities

The cryogenic distillation rare gas recovery system is assumed to be an integral part of the 1500-t/yr BNFP. The two facilities share utilities, laboratories, health physics support, operating personnel, change areas, maintenance areas, warehousing, shops, and offices. The off-gas treatment system has sufficient capacity to handle the total stream from the reprocessing plant. A theoretical floor plan of a facility developed at Idaho National Engineering Laboratory is used in this assessment and is given in Figure 3.3. All normal krypton process operations are handled in the hot cell (located in the upper right-hand corner of the figure). Workers do not normally come in contact with ^{85}Kr.

The reference process design of the treatment facility, shown in Figure 3.4, is a modification of that described in detail by Brown, Knecht and Thomas (1978) and DOE (1979b), and is briefly summarized here. The function of all the equipment outside the cold box is to pretreat the gas stream prior to cryogenic separation. The pretreatment removal of hazardous or troublesome components, such as nitrogen oxide (solidifies NO_2), oxygen (forms radiolytic ozone), and nitrous oxide (causes excessive freezing), is necessary to create a safe worker environment. The purified process gas then enters a heavily insulated and shielded self-contained structure, commonly called a cold box. The cryogenic unit includes a precooler, an absorption-stripping column, and a fractionation still, all shown within the cold box in Figure 3.4. Krypton and xenon are condensed in liquid nitrogen in the adsorption column. The liquid is transferred to the fractionation still, and the nitrogen, krypton, and xenon are separated by fractional distillation. The product that boils off is a mixture of 75% Kr and 25% Xe. The design is for 90% recovery of the ^{85}Kr inventory. We have assumed 95% recovery for this analysis. The materials balance for the modified overall krypton recovery system materials balance is present in Table 3.1.

All volatile fission products other than ^{85}Kr have been removed from the gas before it reaches the rare gas plant. However, ^{85}Kr is present in the gas in large enough quantities to present a radioactive hazard throughout the entire plant. Therefore, the design of the new facilities must take this into account.

Radioactivity of some process streams rises rapidly in the cold box, and the final krypton gaseous product available for ion implantation/sputtering from 30-year decayed fuel will be in excess of 33 Ci/L at 2.7 atm.

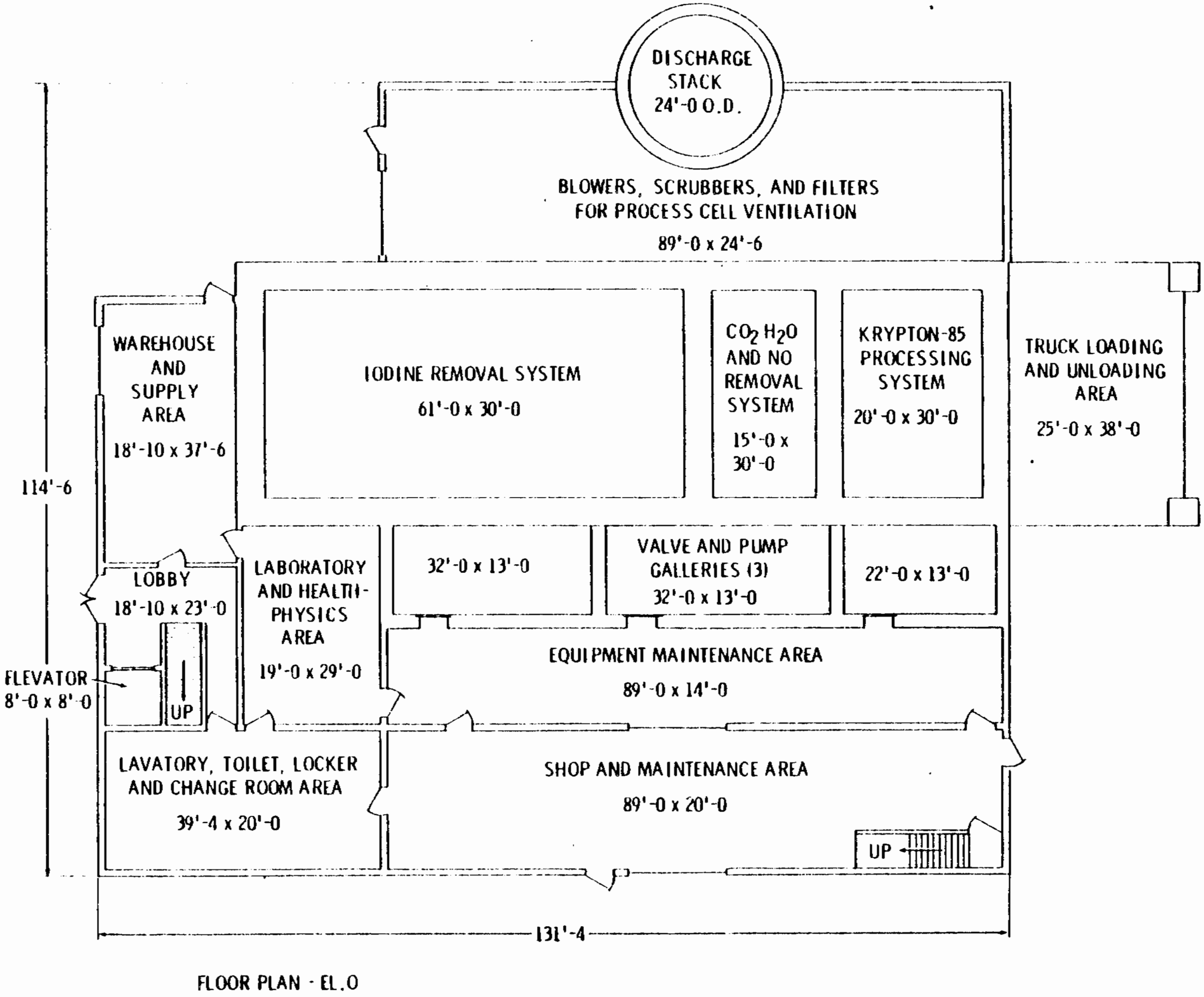


FIGURE 3.3. Dissolver Off-Gas Process Building

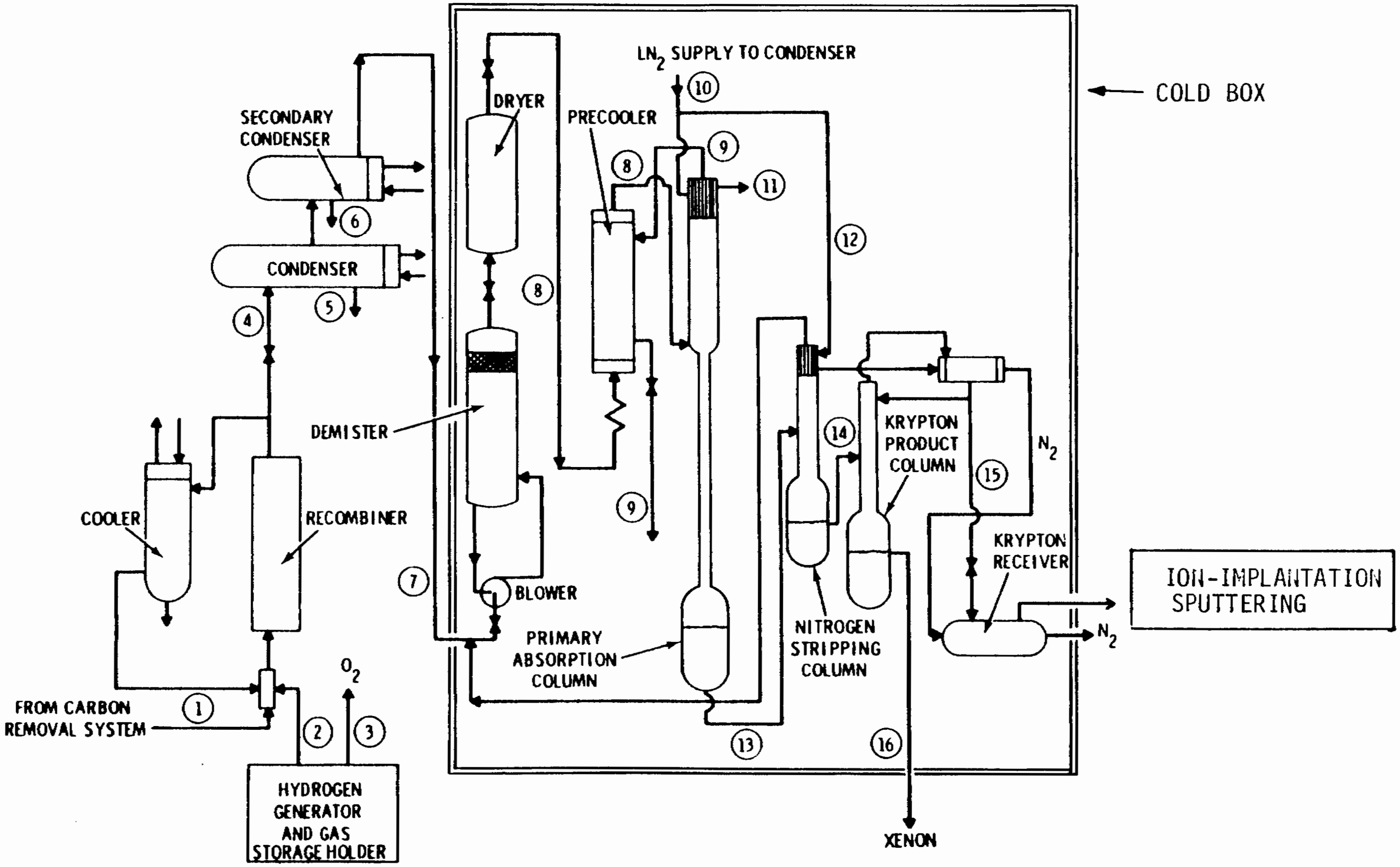


FIGURE 3.4. Process Flowsheet for the Dissolver Off-gas Krypton Recovery System After the Removal of Oxides of Nitrogen and Carbon Dioxide from the Feed Stream (Small circled numbers refer to stream numbers given in Table 3.1)

TABLE 3.1. Material Balance for the Krypton Recovery System (DOE 1979b) (Stream numbers refer to circled numbers in Figure 3.4.)

			Flow by Component, kg/hr												
Stream Number	Description	Total Flow, kg/hr	N_2	O_2	Ar	CO_2	H_2O	NO_x	N_2O	I_2	Br_2	Kr	Xe	H_2
1	Process gas feed (stream 9, Table 4.9.15)	225	171	50.8	2.78	3.8×10^{-4}	5.28×10^{-6}	7.4×10^{-3}	2.81×10^{-1}	6.5×10^{-5}	3.8×10^{-6}	9×10^{-2}	1.3	1.4×10^{-2}
2	Hydrogen production	--	--	--	--	--	--	--	--	--	--	--	--	6.4
3	Waste oxygen	--	--	51.2	--	--	--	--	--	--	--	--	--	--
4	O_2 free process gas	232	172	Trace	2.78	Trace	57.2	Trace	Trace	--	--	9×10^{-2}	1.3	1.1×10^{-1}
5	Condensed H_2O	--	--	--	--	--	50.0	--	--	--	--	--	--	--
6	Condensed H_2O	--	--	--	--	--	6.6	--	--	--	--	--	--	--
7	Low H_2O process gas	232	172	Trace	2.78	--	0.7	Trace	Trace	--	--	9×10^{-2}	1.3	1.1×10^{-1}
8	Dried process gas	232	172	Trace	2.78	--	0.009	Trace	Trace	--	--	9×10^{-2}	1.3	1.1×10^{-1}
9	Cold stripped process gas	221	162	--	2.78	--	--	--	--	--	--	Trace	--	1.1×10^{-1}
10	LN_2 supply total	60	60	--	--	--	--	--	--	--	--	--	--	--
11	Cold nitrogen gas	58	58	--	--	--	--	--	--	--	--	--	--	--
12	LN_2 to N_2 column reflux condenser	2.5	2.5	--	--	--	--	--	--	--	--	--	--	--
13	Feed to N_2 strip column	11.3	9.9	--	--	--	--	--	--	--	--	9×10^{-2}	1.3	--
14	Feed to Kr-Xe fractionation column	1.39	--	--	--	--	--	--	--	--	--	9×10^{-2}	1.3	--
15	Kr product	--	--	--	--	--	--	--	--	--	--	8.1×10^{-2}	2.0×10^{-2}	--
16	Xe stream	--	--	--	--	--	--	--	--	--	--	9×10^{-3}	1.28	--

3.3.2 Shielding and Special Handling

There are several operations within the facility which require special treatment to protect workers from potential hazards. Concrete shielding is generally used in structural walls and lead shielding used where concrete thicknesses for comparable shielding value become cumbersome.

The process gas being treated has an activity of about 2 Ci/m^3[(a)] before ^{85}Kr is separated from it in the precooler and cryogenic section of the plant. The cryogenic distillation hardware can be adequately shielded by 15-cm concrete walls, except for the cold box and the hot cell, which would require about 76 cm of concrete. Limited personnel access to such equipment is possible.

This liquid product contains about 4 Ci/g of ^{85}Kr. One day's production, if stored in a receiver, represents a source of about 7 kCi in a volume of 0.78 L of liquid. This liquid can be stored in dewars for several days without significant loss. If it is desired to hold up the krypton within the cryogenic distillation facility, it can be stored as liquid which can be stored in dewars for several days without significant loss. This source requires a shielding of less than 5 cm of lead inside the cold box shield. The overall cold box krypton inventory (exclusive of the above-mentioned) is estimated to be 7-11 kCi, located almost entirely in the column reboilers and final product column. Shielding of these sources requires less than 10 cm of lead or its equivalent. When the krypton receiver is warmed to approximately $-84^{\circ}C$, the krypton will gasify and flow through shielded transport lines to the ion implantation/sputter facility.

Relative concrete wall thicknesses for the cryogenic facility are shown in Figure 3.3 (Brown, Knecht and Thomas 1978).

(a) This value is a ratio of fuel burnup from ORNL-4451 (ORNL 1970) and ICP-1126 (Brown, Knecht and Thomas 1978) and adjusted by multiplying by 0.144 for 30-year decayed fuel.

3.3.3 Operation and Maintenance

This cryogenic distillation facility is designed to operate 24 hours a day, 7 days a week, for 300 days each year. Normal activities include: monitoring of the NO_x, H_2O, CO_2, and krypton; maintenance; and replacement.

All normal process operations are handled remotely in the hot cell. Compressors and pumps are located in normally unoccupied galleries. Personnel enter the gallery only to install or remove equipment. Where infrequent operations, instrument readings, and some maintenance activities are needed, personnel may be exposed to higher radiation dose rates. In such cases, the dose rate is controlled and limited to less than 1 mrem/hr.

3.3.4 Personnel

Estimates for staffing the krypton recovery system are taken from the DOE (1979a) and presented in Table 3.2 (Details are presented in Table 5.1). These estimates are used to estimate radiation exposures.

3.4 FLUOROCARBON ABSORPTION

The alternative krypton removal system considered for theoretical incorporation into BNFP utilizes fluorocarbon absorption. Fluorocarbon absorption, suggested by Brookhaven National Laboratory (Steiberg 1959), selectively separates krypton and xenon from contaminated off gases. Solubility data have shown that krypton and xenon are much more soluble in dichlorodifluoromethane (refrigerant-12) than are nitrogen, oxygen, and argon. The fluorocarbon absorption process was developed at the Oak Ridge Gaseous Diffusion Plant (Merriman et al. 1980). The primary objective of the fluorocarbon absorption process development program was to provide a detailed design and economic evaluation of a demonstration off-gas decontamination facility. A pilot facility was built specifically to study the reprocessing plant application. Initial process testing gave positive qualitative results. The pilot facility has been operating for several years now using simulated off gas with only trace amounts of ^{85}Kr. Process tests have demonstrated that better than 99.9% of the krypton and xenon can be removed from off gases. In application, a demonstration plant to handle

TABLE 3.2. Staffing Requirements for the Cryogenic Distillation System

Job Classification	Personnel Required man-yr/yr
Operators	5(a)
Radiation Monitors	1
Maintenance Craftsmen	2
Total	8

(a) Operators receive such a low radiation dose from this process, they are not considered further.

the anticipated reprocessing plant off-gas stream would be designed to remove 99% of the krypton and 99.9% of the xenon. The fluorocarbon-based process and peripheral equipment will be defined; optimum operating conditions, including process procedures will be determined; and component and system reliability will be evaluated, including recommendations for necessary support systems. Fluorocarbon absorption is planned for use in the Breeder Reactor Engineering Test (BRET) facility presently to be constructed at DOE Hanford Site.

Serious consideration should be given to the long term use of fluorocarbons. In 1977, the use of fluorocarbons as propellants in aerosol cans was banned, but they are still used as refrigerants, foaming agents, and cleansers. Although a total ban is highly unlikely, continued ozone degradation in the atmosphere could bring about additional regulatory restrictions.

3.4.1 Process and Facilities

The fluorocarbon absorption system would theoretically be expected to occupy a portion of the BNFP as in the case of the cryogenic distillation system (Figure 3.3). The fluorocarbon absorption process would occupy less floor space but would require greater overhead clearance. The reference process design is shown in Figure 3.5.

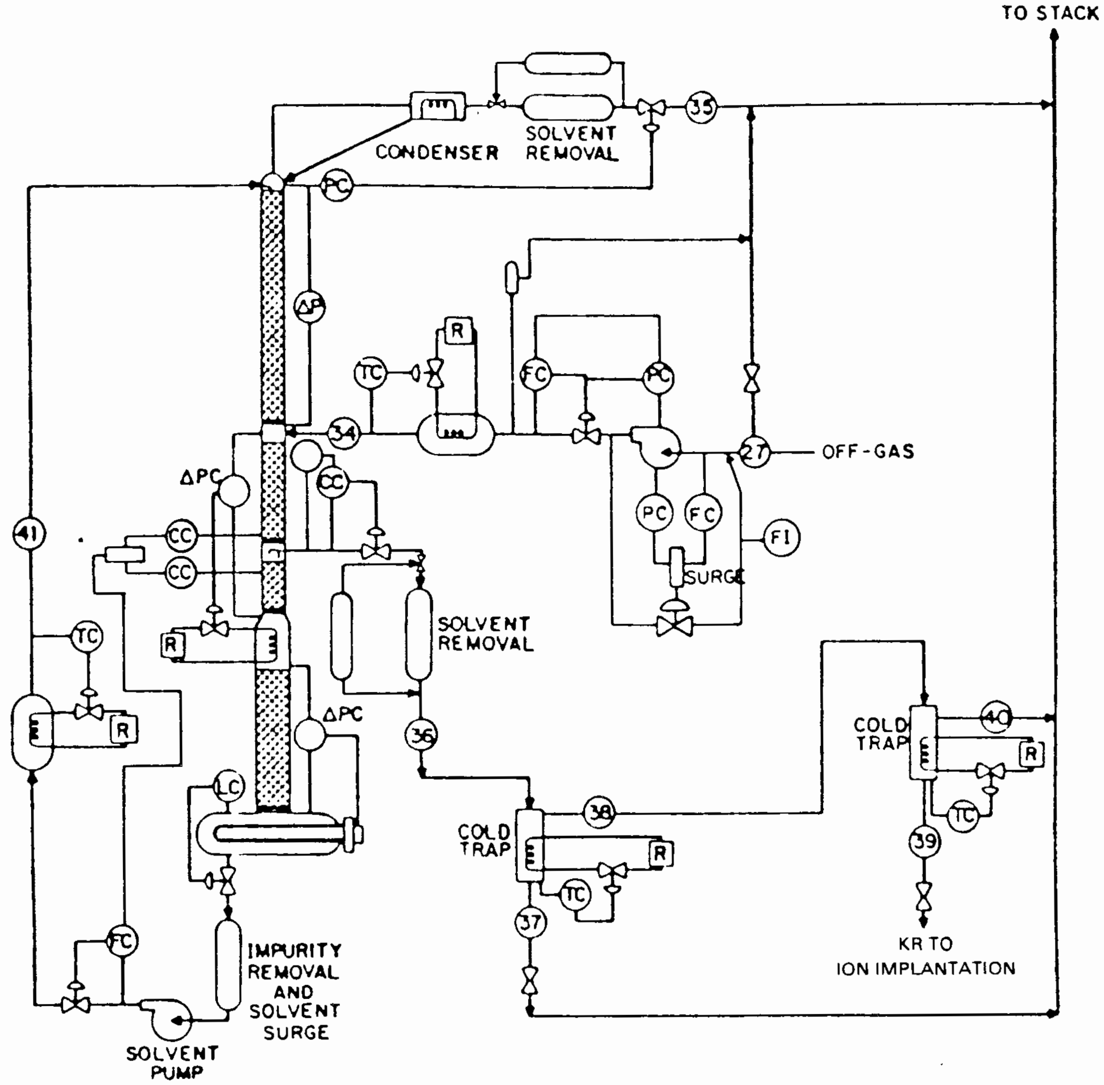


FIGURE 3.5. Fluorocarbon Absorption Process for Krypton Removal (Modified from Merriman et al. 1980) (Small circled numbers refer to stream numbers given in Table 3.3)

Pretreatment of the feed off gas to remove the oxygen or nitrous oxides, as in cryogenic distillation, is unnecessary for this process. The main components of the fluorocarbon absorption process can be contained in a restricted-access area and include a multifunctional absorber/fractionator/stripper column, a solvent purification still, and final product purification equipment. Additional support equipment consists of a feed gas heat exchanger, process gas compressor, solvent pump, solvent cooler, solvent storage tanks, several refrigeration compressors, and a solvent recovery subsystem to remove the solvent vapor from the decontaminated process off gas.

The FRP waste off-gas stream is fed into the krypton recovery column just below the top part, which is the absorber. In this section, the krypton, xenon, and some air are dissolved in the solvent as it flows down the column. The remaining off gas is vented out the top of the absorber, and any solvent vapor is recovered. The middle of the column serves as the fractionator, where the radioactive components dissolved in the solvent are vaporized and collected as a gas side stream consisting of krypton, xenon, air, and solvent vapor. In the bottom part of the column, the bulk of the solvent is stripped of any remaining dissolved gases and recycled back up to the top of the absorber section. A solvent purification still is needed to purge the solvent of water, nitrogen dioxide, and iodine, which stay dissolved and build up in the solvent. The materials balance for the fluorocarbon absorption system for 95% of the ^{85}Kr inventory is presented in Table 3.3.

As the main krypton stream becomes more and more concentrated in the radioactive components, it presents a radioactive hazard within the recovery operation. The krypton column, the solvent recovery system, and the cold traps used to remove the xenon and air from the krypton are all components where krypton will be present in significant quantities and special design considerations will be required.

TABLE 3.3. Material Balance Flowsheet for Krypton Removal by Fluorocarbon Absorption(a) (Waggoner 1982)

STREAM NUMBER:(b)	27	34	35	36	37	38	39	40	41
Pressure, atm	1.0	7.3	1.0	1.0	1.0	1.0	13.6	1.0	7.3
Temperature, ^{0}C	42	-35	20	-15	-20	-150	25	-175	-35
Gas Flow, scfm (ft^3/min)	102.6 (118.3)	102.2 (12.2)	102.1 (109.6)	0.1448 (0.1368)	0.1306 (0.1211)	0.01478 (0.007)	0.01431 (0.001)	0.0006 (0.0002)	-
Liquid Flow, L/day	-	-	-	-	-	-	-	-	74,587
Mass Flow, kg/day	5700	5700	5700	33.4	28.6	2.22	2.2	0.022	112,300
N_2, kg/day	4100	4086	4086	1.8E-2	1.6E-6	1.8E-2	6.5E-4	1.7E-2	-
O_2, kg/day	1219	1219	1219	7.1E-3	7.1E-7	7.1E-3	1.0E-5	7.1E-3	-
Ar, kg/day	67	67	67	8.6E-4	8.6E-7	8.6E-4	4.0E-4	4.0E-4	-
Kr, kg/day	2.16	2.16	<0.0216	2.16	2.1E-5	2.16	2.16	0.0216	-
Xe, kg/day	31.2	31.2	0.	31.2	31.2	0.0312	0.0312	3.12E-4	-
N_2O, kg/day	6-72	0	0	0	0	0	0	0	-
CCl_2F_2 (R-12)	-	-	-	-	-	-	-	-	112,300

(a) Recomputed for an offgas rate of 100 scfm.
(b) Stream numbers refer to circled numbers in Figure 3.5.)

3.4.2 Shielding and Special Handling

Very little special handling is needed for the fluorocarbon absorption removal process.

In the absorption and fractionation process there is, in effect, no holdup within the column. Therefore, the radioactivity in the top two sections (about 0.14 Ci/ft) does not require additional shielding for a restricted-access area where only infrequent entry is required by plant personnel. The standard 15-cm structural concrete walls are more than adequate to keep the dose rates in adjacent occupied areas below the 1 mrem/hr design requirement.

The only stage in the removal process requiring special consideration is the final product purification step. The two cold traps used in this design are shielded with one and one-half inches of lead. The dose rate at the surface of the second cold trap is kept to 5 mrem/hr with the lead shielding, even though the feed stream is 97% krypton. This is acceptable for the restricted-access area containing the process equipment.

3.4.3 Operation and Maintenance

Under normal operating conditions, the column pressure is held constant, and the solvent flow rate is adjusted to produce the maximum krypton concentration in the solvent at the withdrawal location. This is done by using portable gamma scanning; readings are taken above and below the collector to determine the concentration profile and peak. All operations can be handled remotely from a single operating panel.

Routine maintenance will be required for various pieces of equipment. One infrequent operation requires shutting down the cold traps and purging them, and the column can recycle the xenon and krypton internally while this is being carried out. Other equipment, such as pumps and compressors, would be located in a normally unoccupied area where workers would only enter to replace or service a piece of equipment.

3.4.4 Personnel

The fluorocarbon absorption recovery process is expected to have the same labor requirement as the cryogenic distillation process. Thus, the personnel listed in Table 3.2 would apply in this case as well, except that it is anticipated that maintenance requirements would decrease by about a factor of 6.

3.5 ION IMPLANTATION/SPUTTERING

Ion implantation/sputtering provides an effective batch system for immobilizing krypton gas. The process involves the penetration and retention of krypton ions accelerated into a growing sputter-deposited metal film. One major advantage of this method over competing ones is that it can be achieved at low temperature and pressure. Long-term leakage rates of krypton implanted in solids, extrapolated from short-term, high-temperature measurements, appear to be extremely low (Tingey et al. 1979).

Ion implantation/sputtering technology has passed the bench top laboratory scale experimental stage and is now entering the prototype stage. A 1.8 scale-up with remote operations is being developed at Pacific Northwest Laboratory.

3.5.1 Process and Facilities

The ion implantation process involves implanting gases into solids by bombarding the surface of the solid with energetic ions. Positively charged krypton ions are created by collisions between energetic electrons and krypton gas atoms. These positively charged krypton ions, the sputtering ions, are accelerated to the target by applying a large negative voltage to the target. The krypton ions are also accelerated to the substrate to implant in the sputtering deposit by applying a lesser negative voltage to the substrate. A schematic of the process is presented in Figure 3.6.

The process feed gas from the fluorocarbon absorption system is assumed to be 2% xenon, 8% argon and 90% krypton by weight (RMP 1981). The 6% ^{85}Kr in the gas streams becomes 0.86% after 30 years' decay. This gas

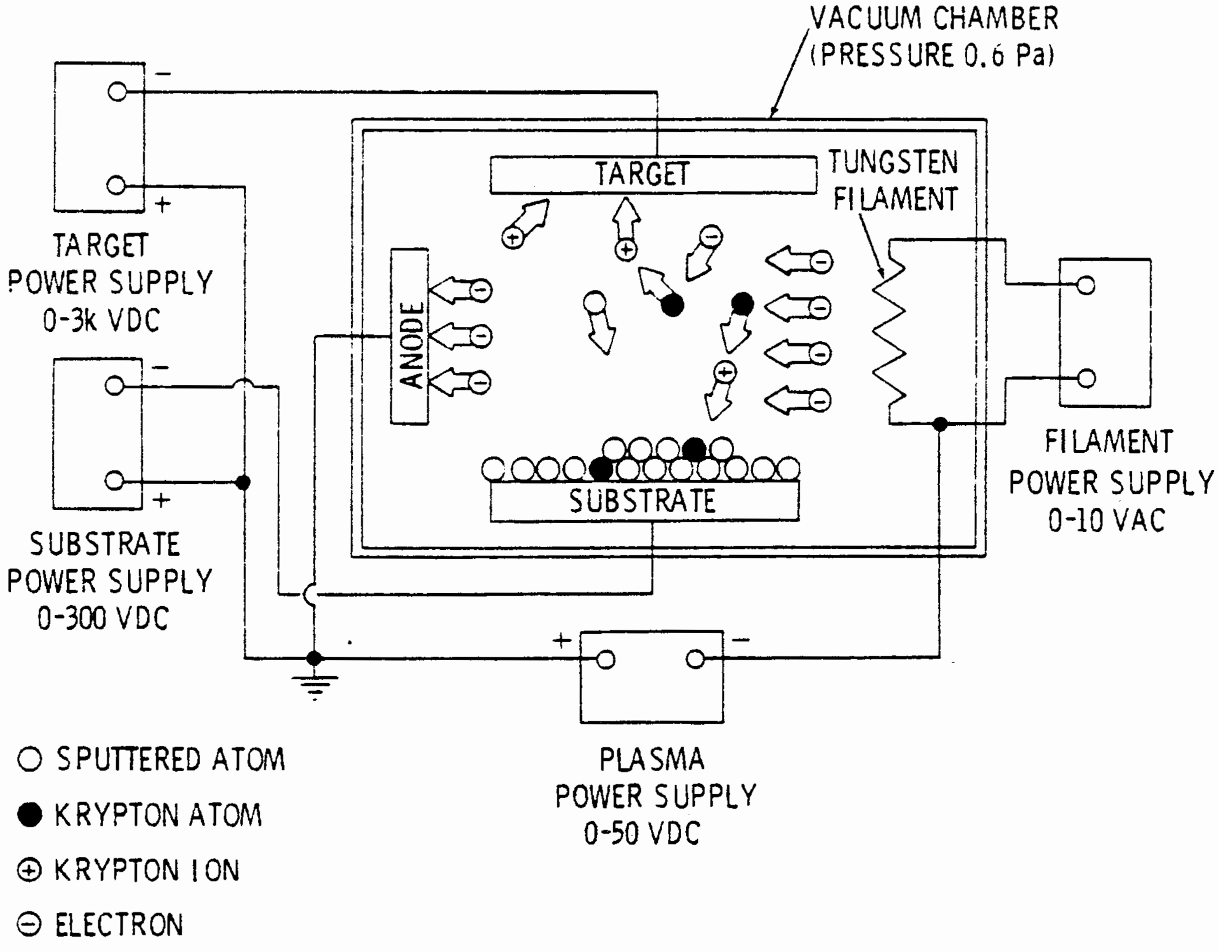


FIGURE 3.6. Thermionically Supported Plasma Sputtering/Ion Implantation Components (Tingey et al. 1979)

composition is the basis for the radiation dose calculations for fluorocarbon absorption and ion implantation/sputtering.

The depth of implantation amounts to only a few atom layers for ion energies in the keV range. To produce large quantities of a solid containing high concentrations of krypton, it is necessary to perform the implantation during film growth by a vapor deposition process such as sputtering where implanted krypton ions are entrapped by the growing sputter deposit. The resulting deposited solid, based on analysis of laboratory scale process deposits, incorporates up to about 13% krypton by weight.

Leakage rates of krypton implanted in solids depend on four diffusion mechanisms discussed by Knecht (1977). Low temperature leakage can be minimized by avoiding structural damage during ion implantation.

The long-term release of ^{85}Kr from two amorphous metal deposits was also investigated as a function of temperature by Tingey et al. (1979) as shown in Table 3.4. These values were extrapolated from short-term, high-temperature measurements. Although long-term tests are needed to confirm these results, it appears that the loss of ^{85}Kr will be negligible at temperatures below 300°C. These data also suggest that packaging this material in additional metal containers may not be necessary. These release rates are so low that storage of ^{85}Kr immobilized in sputtered metals can be accomplished without any significant environmental contamination.

Several matrix materials have been studied and summarized by Tingey et al. (1979). One major advantage of this method over its closest competitor, encapsulation into sodalites, is that it can be achieved at low temperature at low pressure (Bayne et al. 1978).

TABLE 3.4. Fraction of ^{85}Kr Released in 100 Years

	200°C	300°C	400°C
$Fe_{0.79}Y_{0.12}Kr_{0.09}$	1.1×10^{-5}	2.4×10^{-2}	8.5×10^{-1}
$Fe_{0.75}Zr_{0.19}Kr_{0.06}$	7.5×10^{-7}	6.6×10^{-4}	7.1×10^{-2}

The ion implantation/sputtering facility is assumed to be a separate facility tied into the BNFP through a shielded transport pipeline from the cryogenic distillation or fluorocarbon absorption facility. However, this facility shares utilities, laboratories, and health physics support with the FRP. The ion implantation/sputtering facility has sufficient capacity to handle the total krypton gas output of the cryogenic facility. A floor plan of the facility as proposed by Parsons and modified by the authors is shown in Figure 3.7. Modifications to Interim Product Storage were necessary to provide 61-cm concrete shielding to reduce the dose equivalent rate to the design level of 1 mrem/hr as required by ERDA Manual Chapter 6301 (ERDA 1974).

The ion implantation/sputtering facility is a rectangular building with overall dimensions of about 27 m by 15 m. The building is constructed of reinforced concrete for all areas required to survive design basis natural phenomena. Areas required to survive design basis natural phenomena are those areas of the building necessary for containment of ^{85}Kr. Other areas of the building are constructed of reinforced masonry concrete block walls with roofing over metal decking supported by structural steel beams, and are required to survive operating basis natural phenomena.

The facility is on two levels. Power supplies to the process and heating and ventilation equipment are located on the second level, which measures about 27 m by 9 m.

3.5.2 Shielding and Special Handling

Information about operating areas, support areas, and shielding specifications (Figure 3.7 and Table 3.5), along with extensive design information concerning certain functional areas within the plant was used for the occupational radiation dose assessment presented in Section 5.

The original design contained in the preconceptual design (RMP 1981) provided for a 60-day operational supply of feed gas. This design was for the operation of the zeolite absorption process and is unnecessary for the ion implantation/sputtering process, which requires small quantities of

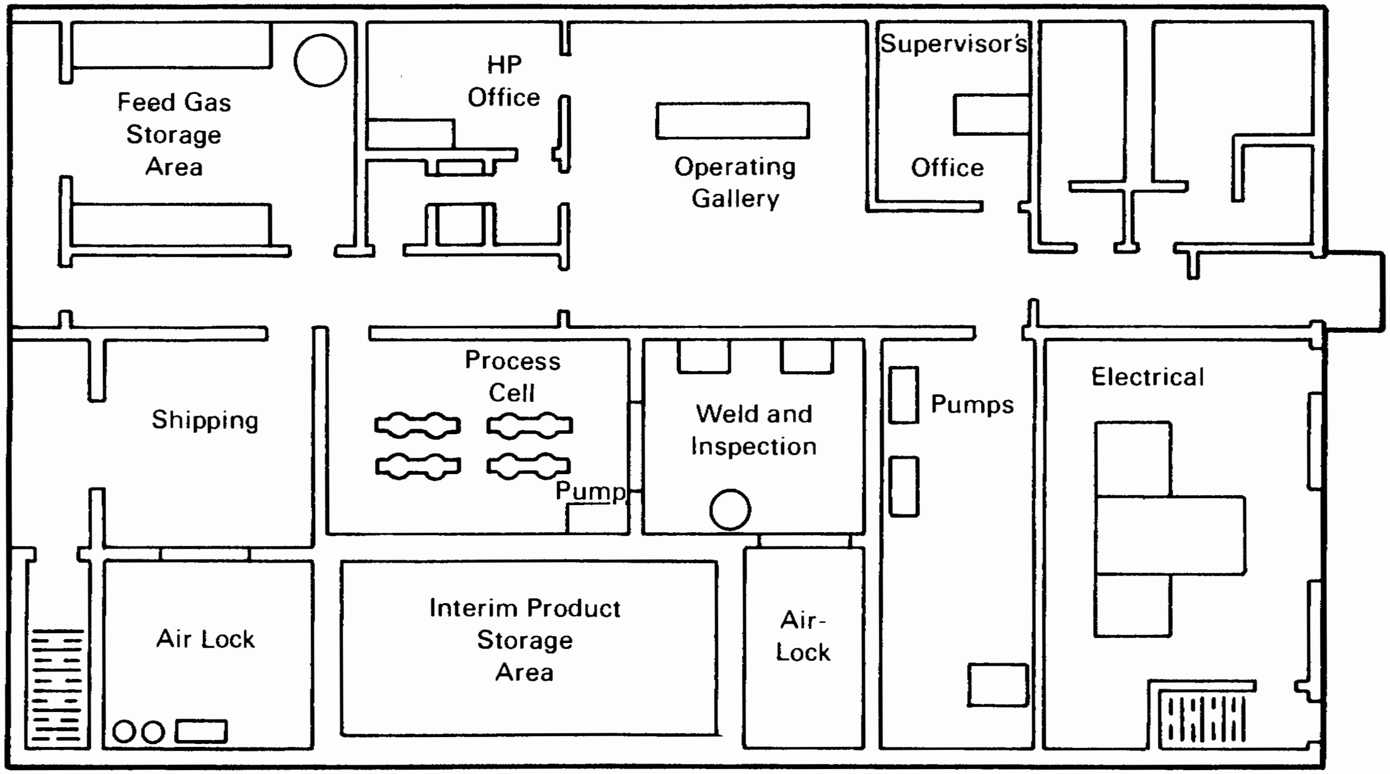


FIGURE 3.7. Modified Floor Plan of Ion Implantation/Sputtering Facility (RMP 1981)

TABLE 3.5. Operating and Support Areas and Shielding Specifications

Facility Area	Area (m^2)	Shielding thickness (cm)	Type of material
Feed Gas Storage Area	20	61 61	reinforced concrete walls & ceiling, concrete floor
Process Cell	43	30 15	reinforced concrete walls & ceiling, concrete floor
Interim Product Storage Area	32	61 61	reinforced concrete walls & ceiling, concrete floor

feed gas at low pressure. Therefore, the lag gas storage area was eliminated and was replaced by a room used for loading canisters for shipment. A single pressure vessel and low-pressure compressor/pump is located in an air lock to collect and store any feed gas that must be purged from other pieces of equipment before contact maintenance is performed.

During normal operations, 47 L/hr of low-pressure feed gas is delivered to the Process Cell for implantation. The krypton encapsulation facility is designed with eight process units (27 cm by 42-cm long) containing a target/substrate assembly. Each unit operates as a batch process, with a calculated processing time of 129 hours, and entraps 34 moles (760 L at standard temperature and pressure (STP) of krypton as the nickel-lanthanum alloy is removed from the substrate and is deposited on the water-cooled target. The deposit weighs about 22 kg and contains some 2.9 kg of entrapped gas. Assuming that the krypton contains 0.86% (30 years' decay) ^{85}Kr, each assembly contains about 9.8×10^3 Ci of ^{85}Kr. Shielded viewing windows are provided for remote operations, substrate inspection, and substrate sealing. An overhead bridge crane services the cell for loading, unloading, and transporting substrate containers. New target/substrate assemblies can be installed in the cell either manually or remotely. The substrate is a standard 25-cm stainless steel pipe and pipe cap, which, when seal welded after processing, provides containment of the encapsulated deposit. The target/substrate assemblies are shielded by 5 cm of lead or equivalent, so that personnel can enter the room in case of equipment malfunction.

Once the krypton is incorporated into the sputtered metal and the substrate/target assemblies are welded shut, the product canister is purged and filled with helium through a connection provided for this purpose. The canisters are leak-checked and then transported to the Interim Product Storage Area. The storage area has the capacity to store encapsulated product captured during one year's operation. The storage rack holds product canisters in a six-by-six horizontal array, to allow cooling by natural convection. The storage rack can accommodate a total of 468 canisters. Inserted product canisters push other canisters in the same position along the length of the storage rack, thereby exposing the oldest canisters at the end of the storage rack. These older canisters are then transported to long-term storage. Shielded hatches are provided for canister maintenance. Because of the radiation levels, the walls of the product storage area must be 61 cm thick to provide adequate shielding for the storage of canisters from one year's operation (if necessary) of the encapsulation facility.

Target, substrate, and filament assemblies are received and stored in the Substrate Storage Area. They are fabricated and tested offsite and shipped to the encapsulation facility as an assembled unit. A total of about 300 assemblies/yr are utilized to encapsulate about 10,000 moles (232,500 L) of feed gas per year. Existing shipping containers do not provide adequate shielding for the krypton activity (9.8×10^3 Ci) expected for canister loading. Containers with increased shielding must be provided; otherwise it will be necessary to reduce the ^{85}Kr inventory per canister and handle more than 1000 canisters per year.

3.5.3 Operation and Maintenance

The krypton encapsulation facility is designed with eight process units. Each unit operates as a batch process, with a calculated processing time as discussed in section 3.5.2. The equipment operates 24 hr/day, 300 days/yr, with a required availability of approximately 70% while the facility is in operation. The facility is designed to have a 30-year life.

3.5.4 Personnel

Estimated staffing requirements for the ion implantation/sputter process are presented in Table 3.6. In addition, other workers will enter the facility for janitorial service.

3.6 STORAGE SCENARIO

In view of the lack of provision for ^{85}Kr in criteria for final emplacement of radioactive wastes, it has been concluded that disposal of the large amounts of ^{85}Kr from FRPs cannot be carried out at the present time (Trevorrow et al. 1983). Therefore, we assume that a large concrete surface retrievable storage facility is colocated with the FRP. The facility is cooled by convective air and therefore requires no regular maintenance schedule. We therefore assume no additional exposure from this retrievable storage facility.

TABLE 3.6. Staffing Requirements for the Ion Implantation/Sputtering Facility

Job Classification	Personnel Required man-yr/yr
Supervisor	1(a)
Operators	2
Radiation Monitor	0.5
Maintenance Craftsmen	1
Total	4.5

(a) The supervisor receives such a low radiation dose from this process that he is not considered further.

4.0 THE BARNWELL NUCLEAR FUEL PLANT ENVIRONMENT

4.1 THE SITE

The BNFP site, located on a 1706-acre tract of land (AGNS 1971), is the representative site for this study. The site is contiguous to the eastern boundary of the DOE's Savannah River Plant (SRP). A site plot plan is shown in Figure 3.2. The land is generally level, with a gently rolling surface, and is heavily wooded with pines and some native hardwood and softwood trees. Portions of the pines were previously harvested. There are no surface streams on the site, and well-defined surface run-off features are essentially absent (Figure 4.1). About 135 acres of the site have been cleared for construction activities. Other site improvements include the 15-acre Beacon cooling water, ecological monitoring, and condition pond for liquid effluent releases to Lower Three Runs Creek. A meteorological tower is located near the southwest corner of the main process area.

Authorities on ^{85}Kr dosimetry agree that the dose from noble gases absorbed in tissue is generally small compared to the dose from direct external radiation and from inhalation (Soldat et al. 1976). Therefore, this analysis requires only information on regional population distribution and site-specific meteorology.

4.2 REGIONAL DEMOGRAPHY

The BNFP is a part of the Barnwell Nuclear Industrial Park. The other nuclear industry currently in the immediate area is Chem-Nuclear Services, Inc., which is engaged in land disposal of low-level radioactive waste and noxious chemicals. Other areas surrounding the site are predominantly rural or restricted as part of the SRP. Agricultural activities are declining. There is a trend toward industrial economy.

The population of Barnwell county has declined in the past two decades. The urban areas with populations greater than 1000 individuals within a 10-mile (16-km) and a 50-mile (80-km) radius of the BNFP are shown in Figures 4.2 and 4.3, respectively. There are no cities within a 25-mile

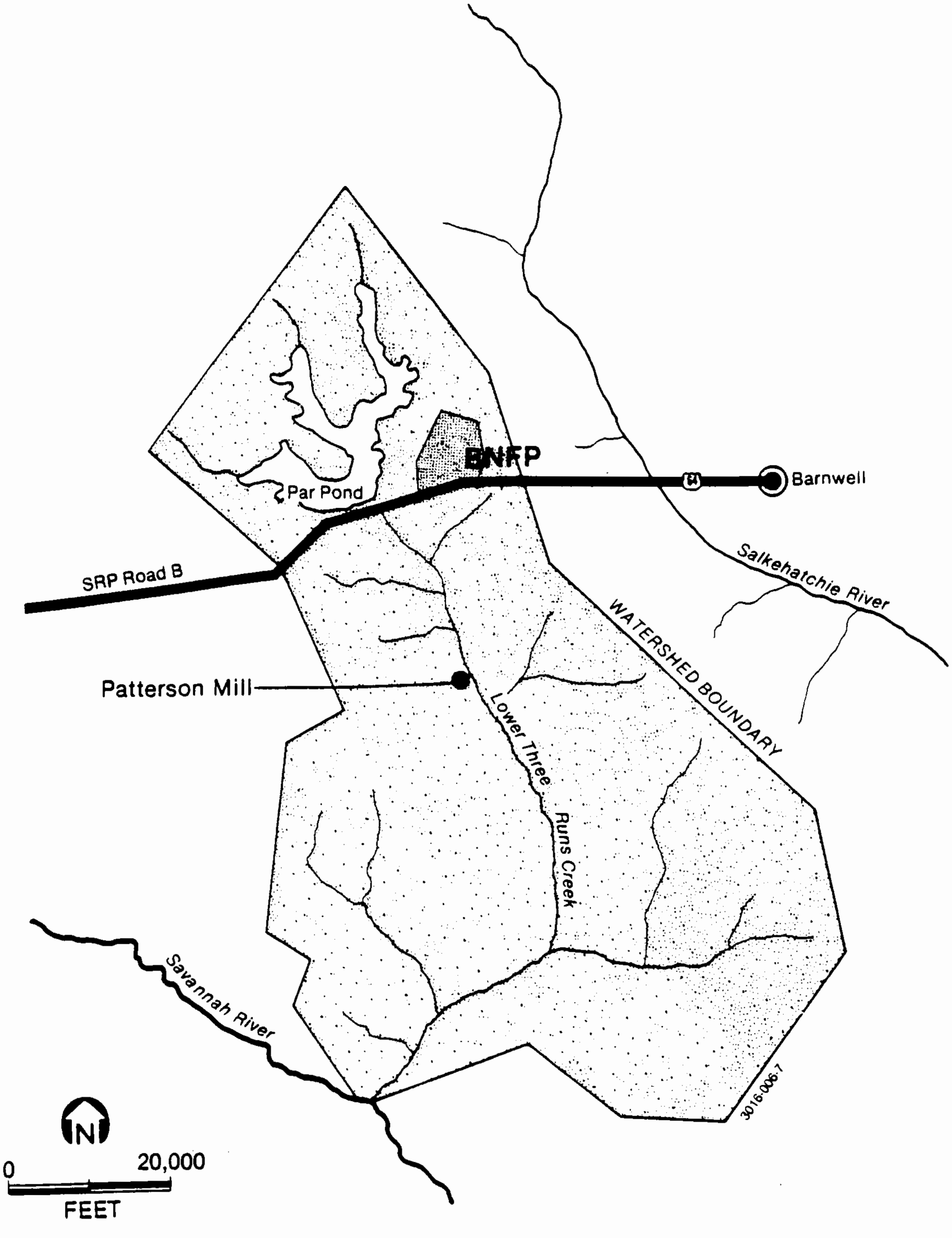


FIGURE 4.1. Lower Three Runs Creek Watershed (NRC 1975)

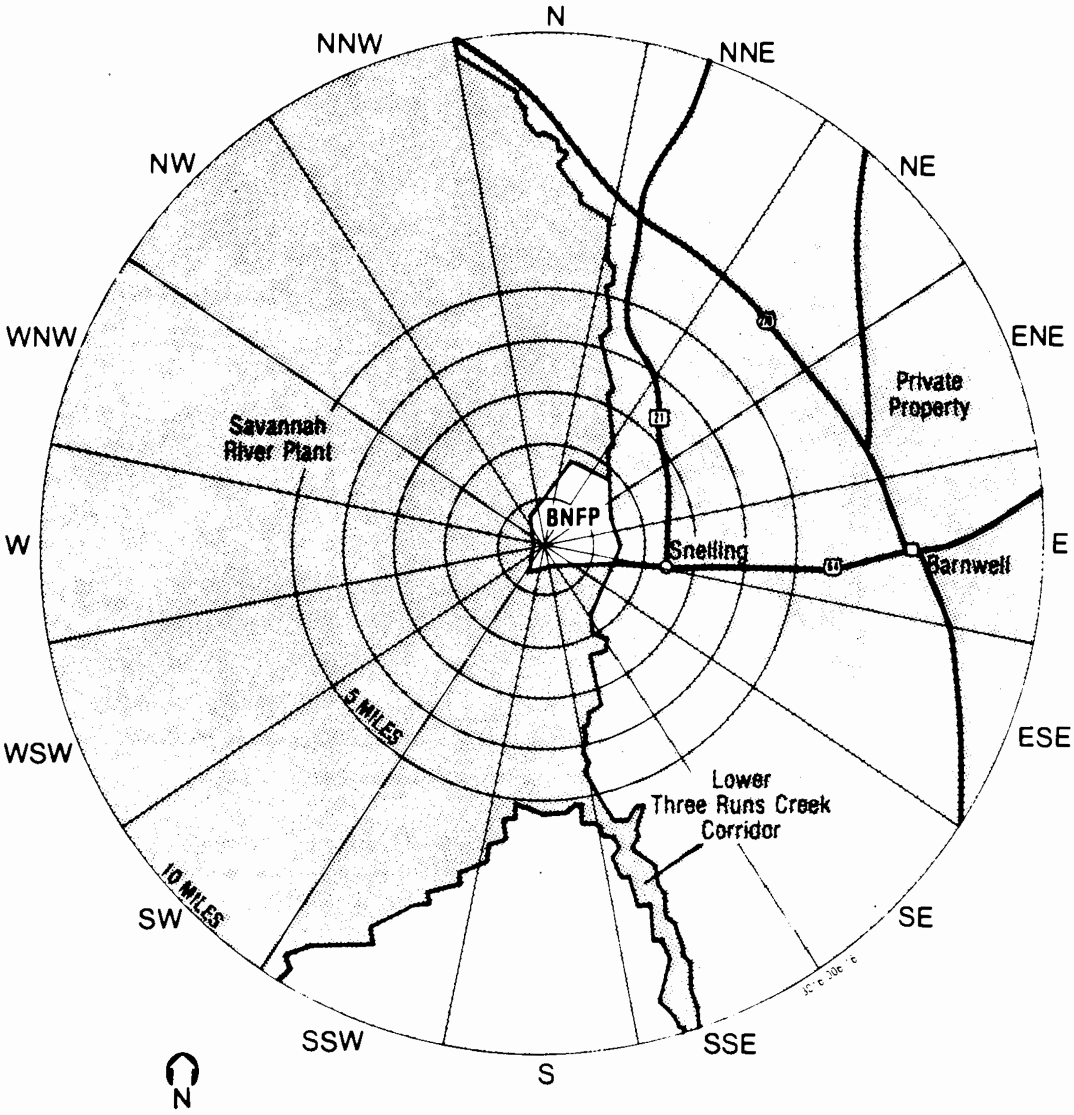


FIGURE 4.2. Sector Detail Within First 10 Miles from BNFP Process Area

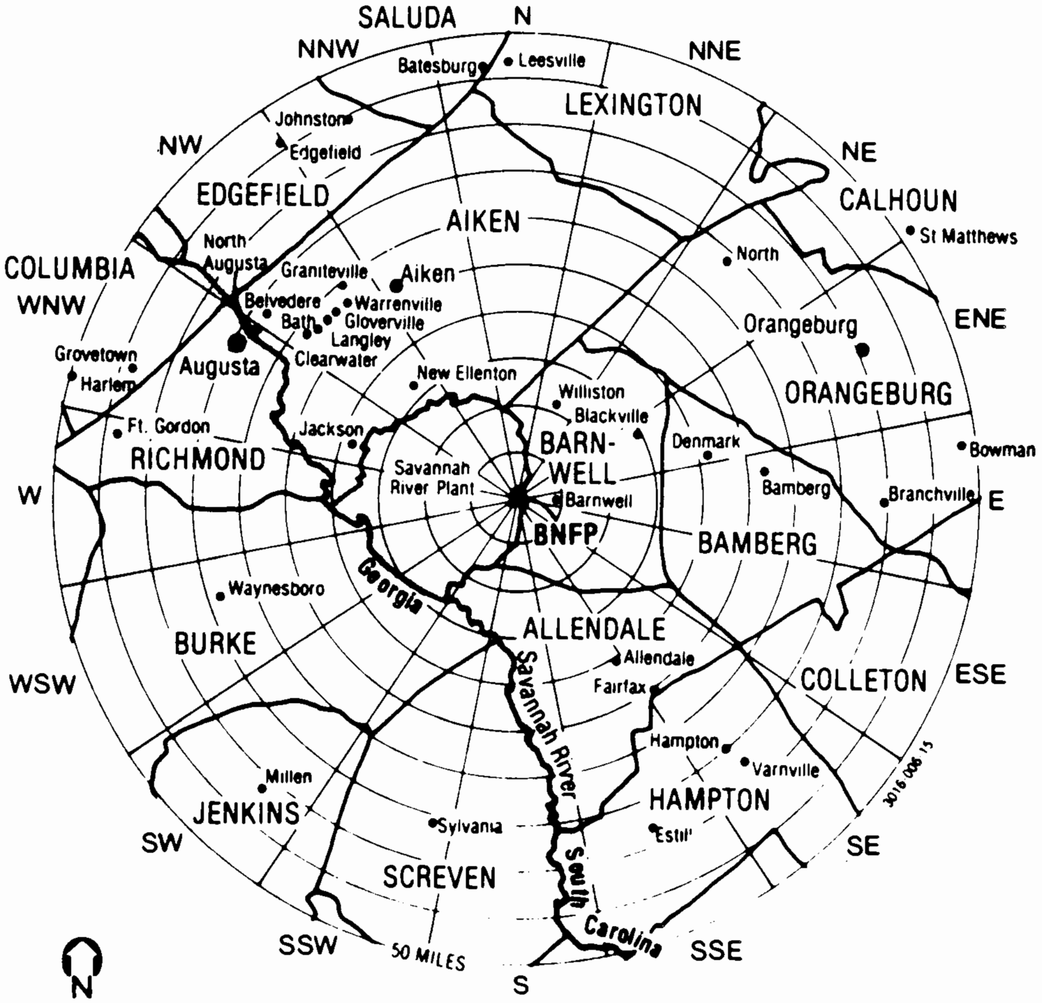


FIGURE 4.3. Urban Areas Greater Than 1000 Population Within 50 Miles of the BNFP

NOTE: See Figure 4.2 for details within first 10 miles radius with a population in excess of 5000. The nearest cities of any size are Aiken, South Carolina (population 16,125), which is about 26 miles NW; Augusta, Georgia (population 59,864), located 33 miles WNW; and Orangeburg, South Carolina (population 13,252), located about 39 miles ENE of the BNFP site. The total 1970 population within 25 miles is 65,209. The projected 1980 population estimate is 79,000. The total 1970 population within a 50-mile radius of the site is 499,000, and the projected 1980 estimate is 658,000. In this scenario, the BNFP is to begin operation in about the year 2000. The projected year 2005 population distribution (Table 4.1) was selected for calculating of the population doses because it was the closest readily available estimate to our operation scenario.

4.3 REGIONAL METEOROLOGY

The BNFP release of process gaseous effluents is from a 100-m stack. There are no hills or valleys in the vicinity that would tend to channel air flow or create mechanical turbulence. There are no bodies of water in the vicinity of sufficient size to create atmospheric diffusion problems associated with a water-land transition zone.

On the basis of the Gaussian distribution equation and the frequency of wind speed and direction for Pasquill categories A through G for the years 1972, 1975, and 1980, annual average atmospheric dispersion factors for airborne material from the 100-m stack were calculated(a) and presented in Table 4.2.

(a) Boone, F. W. 1982. Allied-General Nuclear Services, Barnwell, South Carolina. Memorandum to file, Number ENV C.1, SEC/81/205, Revision 1, January 21.

TABLE 4.1. 2005 Population Distribution Estimates Within 50 Miles of the BNFP (see footnote, p. 4.5)

Direction To	1-2 mi.	2-3 mi.	3-4 mi.	4-5 mi.	5-10 mi.	10-20 mi.	20-30 mi.	30-40 mi.	40-50 mi.	TOTAL
N	0	0	0	0	176	4,188	6,980	9,771	35,395	56,510
NNE	0	0	17	34	352	5,848	6,980	20,123	25,872	59,226
NE	0	0	32	43	352	1,412	7,787	12,741	10,826	33,193
ENE	0	0	32	43	352	12,126	6,000	33,562	12,091	64,206
E	9	17	32	43	7,943	1,924	9,639	10,048	15,889	45,544
ESE	9	342	32	43	352	1,924	3,815	4,921	6,163	17,601
SE	9	17	32	43	352	1,170	5,624	11,508	5,123	23,878
SSE	0	0	26	43	352	7,187	1,660	7,326	5,123	21,717
S	0	0	0	0	352	997	2,079	3,393	4,362	11,183
SSW	0	0	0	0	125	1,315	2,423	8,863	4,362	17,088
SW	0	0	0	0	0	899	2,568	3,093	10,328	16,888
WSW	0	0	0	0	0	450	2,568	13,051	4,622	20,691
W	0	0	0	0	0	814	23,747	33,244	69,400	127,205
WNW	0	0	0	0	0	4,518	6,980	191,570	26,416	229,484
NW	0	0	0	0	0	6,797	50,737	7,257	11,231	76,022
NNW	0	0	0	0	0	4,188	6,980	9,771	11,809	32,748
TOTAL	27	376	203	292	10,708	55,757	146,567	380,242	259,012	853,184

TABLE 4.2. Annual Average Atmospheric Dispersion Factors for the BNFP Separations Facility Main Stack[a]

CHI/Q (SEC/METER CUBED) FOR EACH SEGMENT				SEGMENT BOUNDARIES IN MILES						
DIR.FR.SITE	.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
N	5.987E-09	6.241E-09	7.200E-09	7.498E-09	7.289E-09	5.770E-09	3.556E-09	2.268E-09	1.623E-09	1.247E-09
NNE	9.897E-09	8.738E-09	8.588E-09	8.303E-09	7.798E-09	5.985E-09	3.616E-09	2.295E-09	1.642E-09	1.264E-09
NE	1.474E-08	1.012E-08	9.144E-09	8.550E-09	7.897E-09	5.948E-09	3.535E-09	2.229E-09	1.595E-09	1.230E-09
ENE	1.916E-08	1.211E-08	1.031E-08	9.293E-09	8.388E-09	6.144E-09	3.597E-09	2.267E-09	1.624E-09	1.253E-09
E	2.224E-08	1.346E-08	1.139E-08	1.009E-08	8.972E-09	6.418E-09	3.683E-09	2.301E-09	1.640E-09	1.260E-09
ESE	2.085E-08	1.269E-08	1.107E-08	1.015E-08	9.248E-09	6.889E-09	4.124E-09	2.638E-09	1.903E-09	1.475E-09
SE	1.275E-08	8.378E-09	7.438E-09	6.707E-09	6.014E-09	4.355E-09	2.545E-09	1.613E-09	1.160E-09	8.974E-10
SSE	8.346E-09	6.134E-09	5.301E-09	4.671E-09	4.130E-09	2.916E-09	1.649E-09	1.023E-09	7.266E-10	5.570E-10
S	5.465E-09	5.126E-09	5.037E-09	4.654E-09	4.188E-09	2.984E-09	1.683E-09	1.034E-09	7.271E-10	5.528E-10
SSW	7.301E-09	7.867E-09	8.200E-09	7.687E-09	6.872E-09	4.673E-09	2.430E-09	1.404E-09	9.563E-10	7.130E-10
SW	7.309E-09	9.416E-09	1.060E-08	1.022E-08	9.264E-09	6.425E-09	3.420E-09	1.999E-09	1.366E-09	1.018E-09
WSW	7.913E-09	9.432E-09	1.021E-08	9.817E-09	8.948E-09	6.313E-09	3.429E-09	2.030E-09	1.399E-09	1.049E-09
W	5.970E-09	7.172E-09	8.107E-09	8.106E-09	7.601E-09	5.644E-09	3.265E-09	2.014E-09	1.417E-09	1.077E-09
WNW	5.149E-09	5.769E-09	6.401E-09	6.435E-09	6.117E-09	4.719E-09	2.882E-09	1.846E-09	1.324E-09	1.019E-09
NW	4.760E-09	5.826E-09	6.515E-09	6.533E-09	6.179E-09	4.659E-09	2.732E-09	1.696E-09	1.196E-09	9.105E-10
NNW	4.322E-09	5.497E-09	6.508E-09	6.697E-09	6.411E-09	4.901E-09	2.896E-09	1.799E-09	1.269E-09	9.667E-10

(a) Boone, F. W. 1982. Allied-General Nuclear Services, Barnwell, South Carolina. Memorandum to Files, "Update of Gasious Effluent-Dose Commitment Analysis Documentation," ENV C.1, SEC/81/205 Rev.1. January 21, Table 5.

5.0 KRYPTON-85 DOSE COMMITMENTS

The radiation dose due to ^{85}Kr received by individual members of the public will be reduced proportionately with the recovery percentage of ^{85}Kr at the FRP. However, to recover and immobilize ^{85}Kr, systems have to be purchased and installed, expending capital. In addition, the systems will have to be routinely operated and maintained, thereby expending additional dollars and increasing radiation dose commitment to individuals of the workforce. The recovered ^{85}Kr may subsequently be transported onsite and stored, again resulting in additional cost and additional occupational radiation exposure. The population groups of concern receiving most significant radiation doses are the occupational workforce, the maximum individual in the public sector, and the 50-mile (80-km) population. These trade-offs are now examined individually. A world dose calculation is considered meaningless because the individual radiation dose has been calculated to be less than 0.0007 mrem/yr (Mellinger et al. 1980) for a higher ^{85}Kr release than is being considered here.

5.1 OCCUPATIONAL DOSES FROM ^{85}Kr RECOVERY AND IMMOBILIZATION

Cryogenic distillation and fluorocarbon absorption have been chosen as alternative reference processes for recovery of the BNFP ^{85}Kr. The details of these processes are given in Section 3. The radiation exposures and estimated doses for the workforce from these processes are given here.

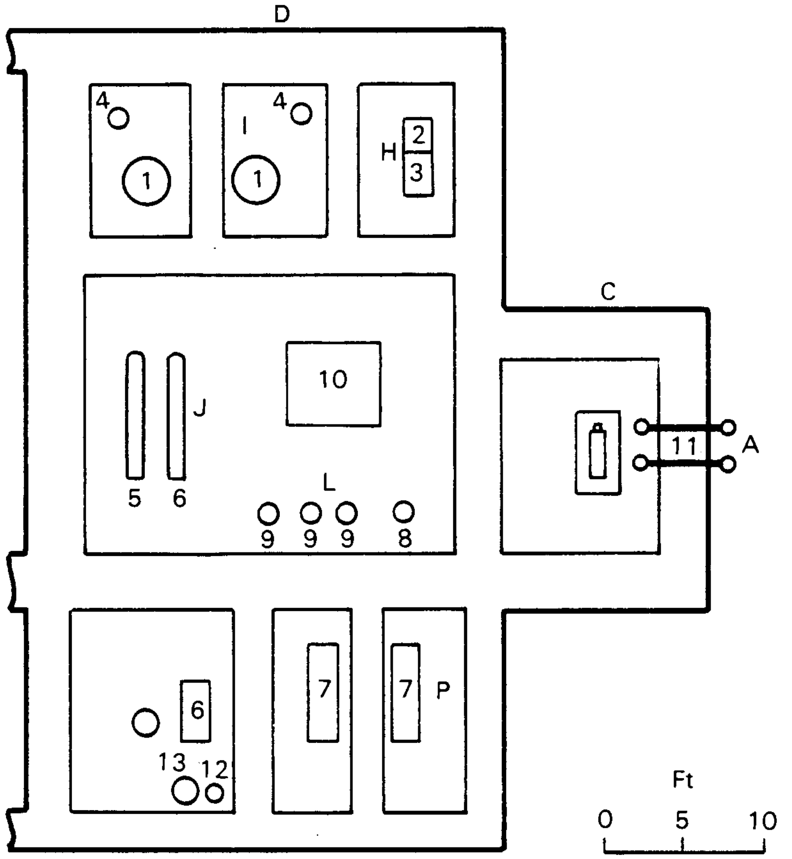
5.1.1 Occupational Doses from Cryogenic Distillation Facility

There are several ways by which workers in the cryogenic distillation facility can receive radiation dose. First, the worker can receive an internal dose from air contaminated with ^{85}Kr. However, it is assumed that negative air balances are maintained, such that a worker receives a negligible inhalation dose. All areas in which the worker may be exposed to ^{85}Kr will be purged before workers enter. If some airborne krypton remains, workers may be required to wear respiratory protection (supplied air) to reduce or eliminate the potential for inhalation of ^{85}Kr.

The second mechanism is the direct exposure to the gamma rays emitted during the decay of ^{85}Kr. The reference cryogenic distillation facility is well-shielded and worker exposures are minimized. The general plan of the DOG system is given in Figure 5.1 along with the dose equivalent rates in mrem/hour at various work locations. Under normal operating conditions, the highest dose rate is only 0.015 mrem/hour through the 1-m-thick concrete walls. It is estimated that a worker would only receive 10 mrem per year working around the shielded cryogenic distillation system. Workforce time estimates are presented in Table 3.2.

The majority of the dose to workers from cryogenic distillation is from maintenance operations on equipment in the hot cells. In order to minimize the workers' exposure, preventative maintenance can be performed on a scheduled basis, after the facility has been shut down and the system has been purged of ^{85}Kr. Breakdown repairs can be handled two ways. The first is to purge the equipment of all ^{85}Kr. Since it requires almost a full day to warm up the cryostat, this method could result in the system being out of commission for two full days or more. The second method is to enter the hot cell and work very rapidly to remove modular equipment so that it can be moved to a shop outside the radiation zone. If ^{85}Kr is not removed from the cold box, the second method can lead to high exposures. To minimize the radiation doses during breakdown maintenance, it is assumed that critical systems, such as compressors, transducers, and liquid level indicators, will have redundant back-up units to minimize down-time.

The equipment used in the krypton recovery facility and the maintenance requirements used for calculating radiation doses are shown in Table 5.1. These maintenance estimates were provided by Pacific Northwest Laboratory (PNL) Maintenance and Craft Services. The ^{85}Kr inventories in each piece of equipment were determined from data taken from Brown, Knecht and Thomas (1978). The resulting dose rates were calculated. Depending upon the assumed amount of shielding around the cold box and whether the cold box is shut down for two days and purged of all ^{85}Kr, the doses can vary from 467 man-rem per year to 0.28 man-rem per year as shown in



Equipment

1. Oxygen Recombiner
2. Hydrogen Generator
3. Hydrogen Storage Gas
4. Primary Gas Cooler (2)
5. Primary Process Gas Condenser
6. Secondary Refrigerated Gas Cooler
7. Process Gas Compressor (2)
8. Separator - Demister
9. Process Gas Dryer (3)
10. Cryogenic Cold Box
11. Manipulators
12. Sump Pump
13. Waste Liquid Receiver Tank

Point	Dose Rate (mrem/hr)
A	0.015
C	0.009
D	0.001
H	0.67
I	0.61
J	0.002*
L	0.004*
P	0.003*

*No ^{85}Kr in Cold Box

FIGURE 5.1. Calculated Dose Equivalent Rates Inside the Cryogenic Distillation Krypton Recovery Facility

TABLE 5.1. Maintenance Requirements for Cryogenic Distillation Facility

Equipment	Number	Man-hr/yr per Piece of Equipment[a]
Valves	91	1
Crane	1	100
Primary process condenser	1	70
Secondary process condenser	1	70
Demister	1	15
Process gas dryer	3	150
Cryogenic cold box		
columns	5	40
receivers	2	15
valves	20	1
electric heaters	9	4
Oxygen recombiner	1	0.5
Hydrogen generator and gas holder	1	52[b]
Recycle gas cooler	2	70
Process gas compressor	2	70
Refrigeration unit	1	100
Diesel engine and electric generator	1	170[b]
Manipulators	6	280
Total System man-hrs/yr		3364[c]

(a) This maintenance time must be multiplied by two (unless otherwise noted) because the work will be done in a radiation zone.

(b) This equipment is outside a radiation zone.

(c) Divide by 2000 man-hr/yr/person to estimate persons required, i.e., 3364÷2000 = 1.7 man-yr/yr.

Table 5.2. Details of this analysis can be found in PNL-3176 (Mellinger et al. 1980). For our purposes, the first case, one-inch Pb on cold box, results in an excessive dose rate and is therefore not considered further.

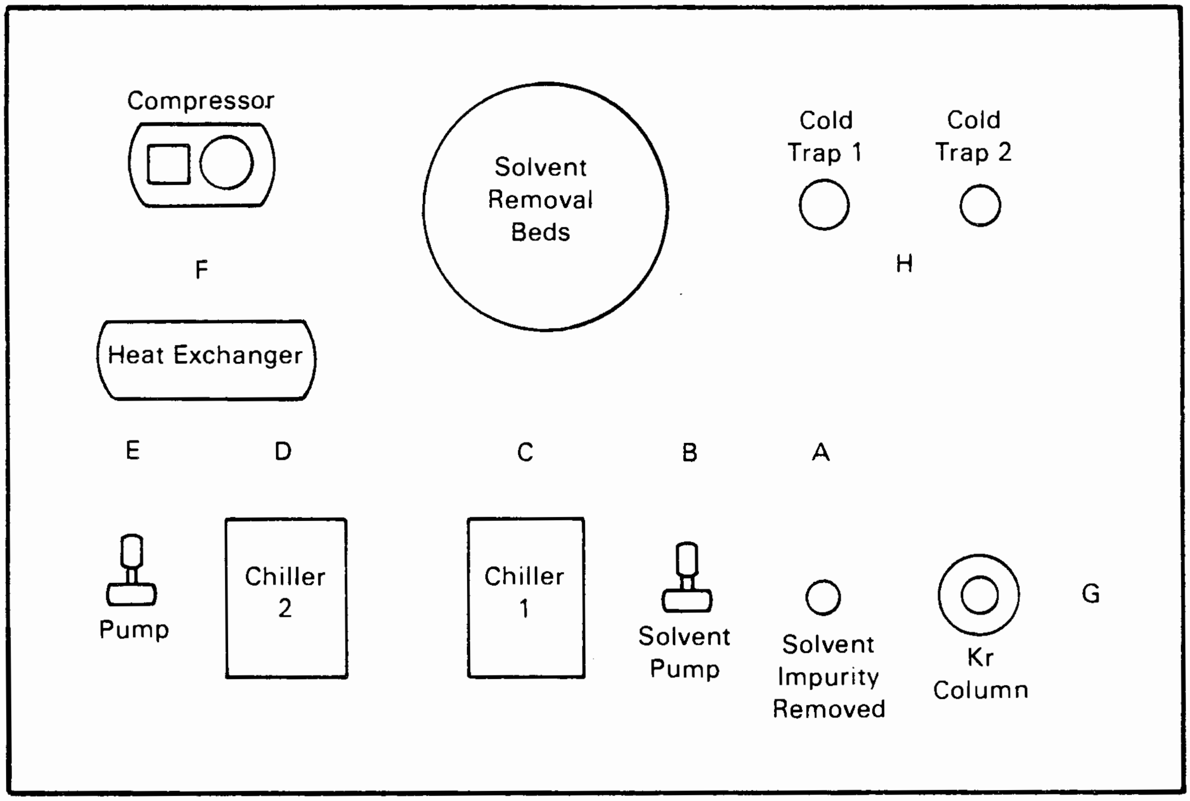
TABLE 5.2. Occupational Exposure for Cryogenic Distillation Facility (man-rem/yr)

Case	Operational	Maintenance	Total
One-inch Pb on cold box	0.01	467	467
Two-inch Pb on cold box	0.01	2.2	2.2
Three-inch Pb on cold box	0.01	0.29	0.30
Cold box system purge for maintenance	0.01	0.27	0.28

5.1.2 Occupational Doses from Fluorocarbon Absorption Facility

The fluorocarbon absorption process and facility have been described in Section 3.4. It is a continuous process, and, compared to the cryogenic distillation process, very little ^{85}Kr is concentrated in the process vessels at one time. Only moderate amounts of shielding are required for the cold traps where the ^{85}Kr is concentrated, and 1-1/2 inches of lead shielding are adequate to reduce the dose equivalent rate to below the design guide of 1 mrem/hr.

All the process equipment can be fitted into the same 6-m-by-9-m process cell used for the cryogenic distillation process. A detailed list of equipment and a process flow diagram is given by Waggoner (1982). No details of the physical layout of the fluorocarbon absorption system are given, so the authors used the equipment listed by Waggoner and devised the layout shown in Figure 5.2. In this scheme, the equipment containing the most ^{85}Kr (the krypton column and the cold traps) is placed at one end of the process cell, and the equipment requiring the most maintenance is placed at the opposite end. Also, 0.6-m to 1.2-m spacing is maintained between each major piece of equipment for ease of maintenance. This layout minimizes radiation exposure, and workers can enter the process cell to perform routine maintenance while the equipment is operating. However, this may not be the way an operating plant is designed.



Location	Dose Equivalent Rate (mrem/hr)
A. Solvent Impurity Removal	0.28
B. Solvent Pump	0.54
C. Chiller No. 1	0.49
D. Chiller No. 2	0.068
E. Pump 2	0.049
F. Compressor/Heat Exchanger	0.041
G. Electric Reboiler/Kr Column	0.69
H. Vent Condenser at Top of Column	0.29

FIGURE 5.2. Dose Rates in Fluorocarbon Absorption Cell Near Various Pieces of Equipment

A modified version of the computer code PERCS (Reece et al. 1984) was used to calculate the dose equivalent rates from the ^{85}Kr in the process cell, assuming that 3.5 Ci of ^{85}Kr are contained in the krypton column, 110 Ci are in Cold Trap Number 1 and 950 Ci are in Cold Trap Number 2, as calculated from the mass flow rates and vessel sizes (Waggoner 1982). These inventories are representative of processing 30-year-old nuclear fuel where the ^{85}Kr content is only 0.86% of the total krypton content. Both cold traps are assumed to have 3.8 cm of lead shielding, as suggested previously (Waggoner 1982).

5.1.2.1 Operational Doses

Assuming the process cell walls (Figure 3.3) are 30 cm of concrete, the dose equivalent rates outside the process cell walls are calculated to be less than 0.005 mrem/hr from the ^{85}Kr inside the process cell. Since the process is designed to be remote and continuous, no one needs to enter the process cell during normal operations. Therefore, the operational doses are assumed to be negligible for normal operations.

5.1.2.2 Maintenance Doses

During routine operation of the fluorocarbon absorption process, the equipment does require periodic maintenance. The maintenance requirements for the equipment in the fluorocarbon absorption process cell are shown in Table 5.3. These estimates, based on previous studies (Mellinger et al. 1980), are time estimates for "generic" pieces of equipment; a particular piece of equipment could require more or less maintenance than shown here.

A modified version of the computer code PERCS (Reece et al. 1984) was used to calculate the dose equivalent rates near each of the major pieces of equipment (Figure 5.2). These dose rates were calculated at a height of 4 feet above the floor (chest height). For the calculations it was assumed that the krypton column was operated in a continuous recycle mode and that both cold traps contained ^{85}Kr. However, for work on the cold traps or on the nearby valves, it was assumed the krypton was vented and the cold traps were purged so that only the ^{85}Kr in the krypton column contributed to exposure. Also, it was assumed that each piece of equipment is isolated

TABLE 5.3. Maintenance Requirements and Integrated Doses for Fluorocarbon Absorption System

Equipment	Man-hours/year Per Piece of Equipment	Integrated Dose (man-rem/yr)
Feed Gas Compressor	70	0.0029
Heat Exchanger	15	0.0006
Vent Condenser	15	0.0044
Krypton Column	40	0.028
Electric Reboiler (1 kW)	4	0.0028
Solvent Removal Bed (6 ft dia x 2 ft)	15(a)	0.015
Solvent Removal Bed (1 ft dia x 2 ft)	15(a)	
Impurities Removal Bed (1 ft dia x 2 ft)	15(a)	0.0044
CO_2 Removal Bed (6 in dia x 1 ft)	15(a)	0.0044
Cold Trap No. 1 (1.5 ft dia x 2 ft, 1.5 in Pb)	15(a)	0.0046
Cold Trap No. 2 (1 ft dia x 2 ft, 1.5 in Pb)	15(a)	
2 Chillers (to -35°C)	70	0.039
14 Valves	1	0.0049
2 Pumps	80	0.047
Total System	550(b)	0.16

(a) Assumed to require the same maintenance as process gas receiver in cryogenic separation process.

(b) Divide by 2000 man-hr/yr/person to estimate persons required, i.e., 550÷2000 = 0.25 man-yr/yr.

and purged of ^{85}Kr before it is worked on, so there would be almost no inhalation dose when the equipment is disassembled.

Using these calculated dose equivalent rates and the annual maintenance time required for each piece of equipment, the occupational doses associated with equipment maintenance were calculated. As shown in Table 5.3, no single item predominates, and the total occupational dose for each year of operation is only 0.16 man-rem. This could be considerably higher if the cold traps are not purged of krypton before workers perform maintenance on the traps or nearby valves. The dose equivalent rate near the cold traps is only 0.15 mrem/hr with the ^{85}Kr purged; with ^{85}Kr in the cold traps the dose equivalent rate can be tens of mrem/hr. Also, if the equipment requiring the most maintenance (compressors, pumps, chillers) were located closer to the cold traps and krypton column, the integrated doses would be much higher. For example, if they were located within 1.2 m of the krypton column on cold traps, the integrated dose for maintenance could be as high as 0.10 man-rem/yr for these items alone.

5.1.3 Occupational Doses from an Ion Implantation/Sputtering Facility

The ion implantation/sputtering process and facility have been described in detail in Section 3.4. This section deals with the occupational doses arising from the operation of the reference ^{85}Kr implantation facility. Estimated manpower requirements for operating the facility are representative of those shown in Table 3.6.

The radiation dose rates at various locations in the facility where personnel will be exposed (capital letters in Figure 5.3) have been calculated (Table 5.4) using a slightly modified version of the computer code PERCS (Reece et al. 1984), and ISOSHLD (Engel et al. 1966 and Simmons et al. 1967) for gammas and bremsstrahlung contribution, respectively. The contribution from bremsstrahlung is only important for lightly shielded sources; the low energy photons do not penetrate the massive shields used in the ^{85}Kr encapsulation facility. The computer codes PERCS and ISOSHLD calculated similar dose rates for the heavily shielded ^{85}Kr sources (behind concrete or lead). Because PERCS is much easier to use, it was used for most of the dose rate calculations.

As noted in the section describing the ion implantation/sputtering facility, adequate shielding was not provided in the preconceptual reference design, so the PERCS code was used to calculate additional shielding requirements. Areas that required additional shielding were the sputtering stations, the canister welding and inspection stations (5 cm of lead shielding added), and the canister storage (61 cm of concrete). Minimal

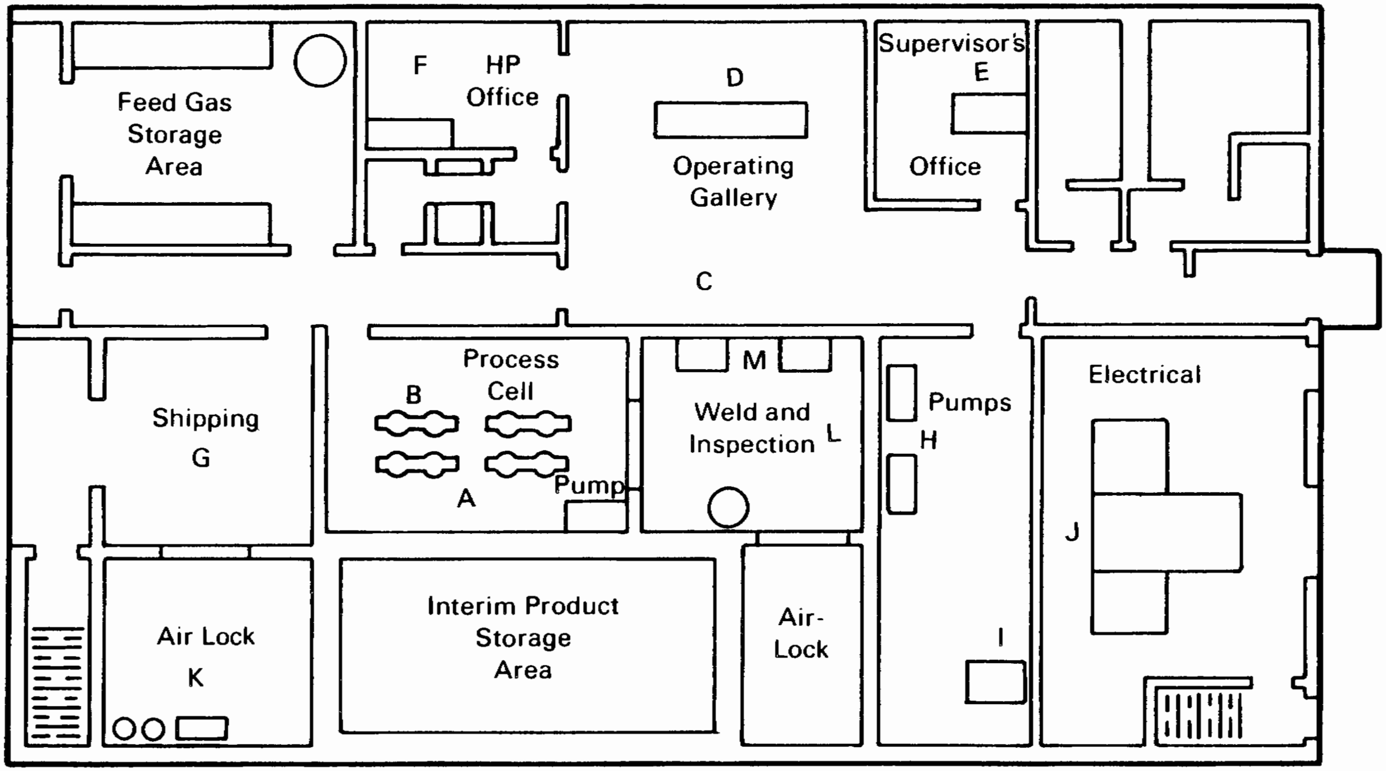


FIGURE 5.3. Locations Where Dose Equivalent Rates are Calculated for Ion Implantation/Sputtering Facility (See Table 5.5 for doses)

TABLE 5.4. Calculated Dose Equivalent Rates at Various Locations Inside the Ion Implantation/Sputtering Facility

Location	Description	Dose Equivalent Rate (mrem/hour)
A	Process cell near product storage	1.5
B	Process cell near sputtering apparatus	1.6
C	Canister welding in operating gallery	0.06
D	Operating gallery console	0.1
E	Supervisor's office	0.0004
F	Health physics office	0.1
G	Shipping	0.5
H	Pumps	0.09
I	Pumps	0.01
J	Electrical equipment	0.001
K	Compressor in air lock	3.5(a)
L	Bridge crane	0.1
M	Between welder and leak detector	0.9

(a) It is assumed that the compressor should not be located outside the seismically reinforced structure; i.e., it should be bounded by 30-cm-thick walls.

shielding was added so that the facility could be licensed without extensive changes in the design or layout.

The two main sources of exposure to workers in the facility will be examined in detail: direct exposure to radiation from ^{85}Kr in the facility during normal operations and exposure during routine maintenance or repair on equipment in the facility.

5.1.3.1 Occupational Doses from Normal Operations

The operational doses received in processing the ^{85}Kr are summarized in Table 5.5. Four persons are in low-level radiation fields while in the building for 2000 working hours per year and while in the sputtering cell to set up and remove target/substrate assemblies for immobilizing ^{85}Kr. A significant contribution comes from preparing canisters for shipment where

TABLE 5.5. Operational Doses from Handling and Processing ^{85}Kr in the Ion Implantation/Sputtering Facility

Source of Exposure	Annual Integrated Dose (man-rem/yr)
General building background	0.31
Radiation surveys	0.30
Entry into process cell to set up sputtering assembly and stations	0.10
Welding on lids and inspecting canisters for leaks	0.20(a)
Preparing canisters for shipment	0.27
Total	1.2 man-rem/yr

(a) Assumes no lead shielding around canister during welding operation; this value is reduced to 0.02 man-rem/yr with addition of 5 cm of lead shielding.

personnel are in high radiation areas to load, inspect, and survey the shipping containers for shipment to a permanent storage facility, assumed to be a retrievable surface storage facility or a geologic repository. This dose is significant because two persons must come in close proximity to each canister in the shipping container to verify radiation levels and the integrity of the shipping container. One of these persons is the health physics technician who also has other functions to perform in this facility. It is assumed that radiation surveys are performed whenever a worker enters a radiation zone for normal operation or for maintenance as well as routine surveys around the facility. The total integrated dose for normal operations is estimated to be about 1.2 man-rem per year of operation. This could be reduced considerably by redesigning the facility to include more shielding and remote operations, but the maintenance doses would then increase.

5.1.3.2 Occupational Doses from Equipment Maintenance

A significant occupational dose is received by workers who service and maintain the equipment in the facility. A complete list of all the equipment in the facility is given in a document prepared by The Ralph M. Parsons Company (RMP 1981). Table 5.6 summarizes the most important pieces of equipment and lists estimated maintenance times that are taken from previous studies (Mellinger et al. 1980). Using these estimated times and the radiation dose rates calculated at various locations by the computer code PERCS, the radiation doses involved in equipment maintenance in the facility during normal operation are estimated. None of these integrated doses are excessive, even though some of the equipment is located in potentially high radiation fields. Care must be taken in designing the shielding around the feed gas compressors to avoid excessively high dose rates in the lag storage area. A problem might also arise from entrainment of ^{85}Kr in the vacuum lines. The location of the equipment is shown in Figure 5.3. The total integrated dose for normal contact maintenance operations was about 1.4 man-rem per year. Depending upon the location of the vacuum pumps and where maintenance is performed, the calculated values range from 0.7 to 1.5 man-rem/yr. This could be considerably higher if problems developed in the interim product storage area. It would be necessary to remove the ^{85}Kr to gain access to these high radiation areas. Another area that must be considered is the piping containing feed-gas. Unshielded pipe runs could result in very high dose rates. Although it is not shown in the preconceptual design, it would be a good idea to design a well shielded pipe gallery.

5.1.4 Summary of Occupational Doses

The total external integrated doses involved in the capture of ^{85}Kr each year can be determined by adding together the external doses from the cryogenic distillation facility or the fluorocarbon absorption facility with the external doses from ion implantation/sputtering facility. These doses are summarized in Table 5.7. Since the facility is operated over a 30-year period, these integrated doses can be multiplied by 30 to calculate the occupational dose over the lifetime of the plant.

TABLE 5.6. Maintenance Requirements and Occupational Doses from Equipment in Radiation Zones in the Ion Implantation/Sputtering Facility

Equipment	Number	Man-hours/year per piece of equipment	Integrated Dose (man-rem/yr)
Feed gas compressor	1	70	0.24
Vacuum pumps for sputtering chambers	2	60	0.18
Other pumps in facility	-	--	0.016
Pressurized feed gas filter	1	unknown	--
Sputtering heads and shielded cooling jackets	8	20	0.24
Mini-trailer robo-carrier or remote stacker/retriever	2	100	0.16
In cell bridge crane	1	100	0.01
Remote welder in cell (replace with spare)	1	10	0.009
Helium leak detection (replace with spare)	1	10	
Master-slave manipulators	4	280	0.38
Valves in process lines and radiation zones	122	1	0.12
Total System		1912(a)	1.36

(a) Divide by 2000 man-hr/yr/person to estimate persons required, 1912÷2000 = 0.96

TABLE 5.7. Summary of Operational and Contact Maintenance Doses for Each Year of Operation of ^{85}Kr Separation and Encapsulation Facilities

	Integrated Dose (man-rem/yr)		
	Operational	Maintenance	Total
Cryogenic Distillation recovery facility	0.01	2.2 to 0.27	2.2 to 0.28
Fluorocarbon Absorption recovery facility	small	0.16	0.16
Ion Implantation/Sputtering encapsulation facility	1.2	1.4	2.6
Total (Cryogenic + Ion)	1.2	3.6 to 1.8	4.8 to 3.0
Total (Absorption + Ion)	1.2	1.6	2.8

5.2 PUBLIC EXPOSURE TO ^{85}Kr RELEASED ROUTINELY FROM THE BNFP

When ^{85}Kr is routinely released to the atmosphere directly from the DOG, the occupational workforce is not generally subjected to its radioactivity in the workplace. However, members of the workforce are exposed about 16 hours a day and the surrounding population is exposed full-time to the ^{85}Kr released to the environment. Therefore, the total release scenario (no recovery) is used to calculate doses for the maximum individual and the 50-mile regional population. The dosimetry of ^{85}Kr has been presented elsewhere (Mellinger et al. 1980). Equations used for these dose calculations are presented in Appendix B.

5.2.1 Maximum Individual Dose

The maximum ground-level concentration of ^{85}Kr occurred 1-2 miles east of the site. The 30-year collective radiation dose commitment to a hypothetical individual continually located at this point during continuous routine release was about 0.7 mrem to the total body, 1.3 mrem to the lung,(a) and about 50 mrem to the skin.

(a) Inhalation dose plus total-body dose.

For ^{85}Kr with relatively little penetrating (gamma) radiation, the critical organ will be the skin, even considering its less restrictive dose standard. However, where collective doses are calculated, the total-body dose is preferred for estimating health effects.

The annual dose rate to the total body of the maximum individual was about 0.02 mrem. Proposed revisions of 10 CFR 20(a) include lower (de minimis) limits, below which radiation dose levels need not be considered in dose assessments. The de minimis limit proposed for individuals is 1 mrem/yr. Krypton-85 doses cannot be compared solely to the 1 mrem/yr because both tritium and ^{14}C, also released by an FRP, would deliver larger radiation doses than ^{85}Kr. However, this quantification of a de minimis dose is a step in the right direction.

5.2.2 Population Dose

The population values used in this calculation (Table 4.1) are estimates for the year 2005.(b) The annual average meteorological dispersion factors presented in Table 4.2 are used as shown in Appendix C to estimate radiation doses to the public.

The population is distributed relatively evenly beyond the influence of the DOE SRP site. The largest single population center within the 50-mile radius of the BNFP is Augusta, Georgia, located about 35 miles WNW.

The 30-year cumulative 50-mile population dose commitment was about 70 man-rem. The annual population dose was calculated for the 50-mile population (850,000 people) to be about 2.35 man-rem/yr. This averages out to be a per-capita dose of about 0.003 mrem/yr. The proposed revisions of 10 CFR 20 include lower limits of collective doses to individual members of a general population of 0.1 mrem/yr. However, again, these are not directly comparable.

(a) An NRC proposed draft for changes in 10 CFR 20-(Subpart E De Minimis Levels for doses to Individual Members of the Public) September 26, 1983.

(b) Boone, F. W. 1982. Allied-General Nuclear Services, Memorandum to Files, "Update of Gaseous Effluent-Dose Commitment Analysis Documentation," ENV C.1, SEC/81/205, Rev 1, January 21.

5.2.3 Summary of Estimated Occupational Doses vs. Public Radiation Doses

It has been shown in Table 5.8 that if ^{85}Kr is captured by cryogenic distillation and immobilized by ion implantation/sputtering, the annual average man-hours requires 7 men full-time or more than 7 men part-time. We assume a range of 7-9 men for cryogenic distillation and ion implantation/sputtering and 5-7 men for fluorocarbon absorption and ion implantation/sputtering. The average annual per-capita dose to the 7-9 exposed workers was estimated to be about 570-450 mrem (4 man-rem/yr ÷ 7 men). If fluorocarbon absorption is used instead of cryogenic distillation, the average annual per-capita dose to the 5-7 workers was estimated to be about 600-425 mrem (3 man-rem/yr ÷ 5 men).

Comparing the average occupational exposures of the two recovery options coupled to ion implantation/sputtering and immobilization (about 3.5 man-rem/yr) with the 50-mile population dose (2.3 man-rem/yr), the dose commitment to the occupationally exposed workforce was not significantly higher than for the public. However, the annual average per capita dose to

TABLE 5.8. Comparison of Whole-Body Radiation Dose Rates from Two Krypton Removal Technologies with Ion Implantation Versus Public Exposure from Routine Release to the Environment[(a)]

	Man-rem/yr	Man-yr/yr	Men	Average Individual mrem/yr
^{85}Kr Recovery and Immobilization				
Cryogenic + Ion	4	7	7-9	570-450
Absorption + Ion	3	5	5-7	600-420
Routine Total ^{85}Kr Release				
Maximum Individual	0.00002	---	1	0.02
50-mile Population	2.4	---	850,000	0.003

(a) This comparison is based on the release of 3×10^6 Ci ^{85}Kr during processing of 1500 t/yr of 30-yr aged fuel.

the occupationally exposed individual was about a factor of 1.7×10^5 greater than to an individual member of the public (500 mrem vs. 0.003 mrem). In addition, as noted in Chapter 3, only 95% of the ^{85}Kr is assumed to be recovered. Therefore, in the recovery case, a small dose is also delivered to the public from the 5% ^{85}Kr still released to the environment, and dose is also delivered to workers from any onsite transport of waste containers that may occur. These doses tend to balance out and are not considered further. Therefore, there appears to be a possibility of increasing human health effects if ^{85}Kr is captured, concentrated, and stored rather than routinely released to the environment.

6.0 HUMAN HEALTH EFFECTS

Health effects likely to result from exposure to low levels of ionizing radiation have been discussed in the BEIR III (Biological Effects of Ionizing Radiation) issued by the National Academy of Sciences (NAS 1980). Cancer mortality due to radiation exposure was reviewed, and a model for quantifying risks was developed. Problems that were encountered in relating estimates of risks due to exposure to low levels of radiation are discussed in detail throughout the report, and are briefly summarized below (BEIR III, pp. 142-143).

> "The quantitative estimation of the carcinogenic risk of low-dose, low-LET (linear energy transfer) radiation is subject to numerous uncertainties. The greatest of these concerns the shape of the dose-response curve. Others pertain to the length of the latent period, the relative biological effectiveness (RBE) for fast neutrons and alpha radiation relative to gamma and x radiation, the period during which the radiation risk is expressed, the model used in projecting risk beyond the period of observation, the effect of dose rate or dose fractionation, and the influence of differences in the natural incidence of specific forms of cancer. In addition, uncertainties are introduced by the characteristics of the human experience drawn on for the basic risk factors, e.g., the effect of age at irradiation, the influence of any disease for which the radiation was given therapeutically, and the influence of length of follow-up."

One of the more severe problems is that in populations where the dose and exposure rates are low, extremely large sample sizes are required to reliably quantify the magnitude of effects. In the judgment of the BEIR III committee, none of the studies of human populations that have been exposed primarily at low levels provide sufficient information for risk estimation. Thus, it is necessary to extrapolate from estimates based on data from populations which include persons exposed at relatively high doses and dose rates such as the Japanese atomic bomb survivors and British ankylosing spondylitis patients who were medically treated with radiation.

Although estimates of risks due to radiation have often been based on a linear extrapolation model (NAS 1972; UNSCEAR 1977; ICRP 1977), the BEIR III committee adopted a linear-quadratic function as providing the most plausible description of the dose-response relationship in the low-to-intermediate range. BEIR III also provided alternative estimates based on the linear and pure quadratic models. The use of a model (such as the linear-quadratic) that provides for a reduction in linear effects with low Linear Energy Transfer (LET) radiation for reduced doses and dose rates can be justified based on evidence that is summarized in a report of the National Council on Radiation Protection and Measurements (NCRP 1980). In this report it is stated that it is clear from the data obtained from all endpoints examined, from cell death to tumor induction, that a reduction in dose rate in general results in a reduced biological effect.

Another difficulty in obtaining lifetime risk estimates is that none of the populations on which estimates of effects are based have yet been followed to the end of their lifespans. Two approaches were used by the BEIR III committee to extend risk estimates beyond the period represented by follow-up data. With the absolute risk model, it is assumed that the number of excess cases per unit of population per unit of time and per unit of radiation dose remains constant over a specified time period. With the relative risk model, it is assumed that the ratio of the excess cancer risk to the spontaneous age-specific risk remains constant over the specified period. Since spontaneous rates generally increase with age, the relative risk model will yield larger numbers for the years beyond the follow-up period. The most recent data on Japanese survivors (Kato and Schull 1982) as well as data on British ankylosing spondylitis patients (Smith and Doll 1982) indicate that the relative risk projection model is probably more appropriate than the absolute model.

We compared estimates of cancer mortality based on the results presented in the last two columns of Table 5.8. For this purpose, we used

BEIR III estimates for a single exposure to 10 rad.(a) In all cases examined in this study, dose rates are lower than those considered by BEIR III, especially the situation involving total ^{85}Kr release to the environment. To obtain estimates for the number of cancers expected in each of these situations we have reduced the BEIR III 10-rad estimates of risk in proportion to the estimated ^{85}Kr dose, and then multiplied by the number of individuals at risk to obtain the expected number of cancer deaths induced by the radiation received in a single year of plant operation. These estimates are then multiplied by 30 and the exposed population to obtain estimates of the total cancer mortality expected to result from 30 years of plant operation as presented in Table 6.1. Estimates based on several models are presented in BEIR III, but here we consider only the estimates based on the linear-quadratic relative risk model(b) (2255 deaths/10^6 people/yr for a 10-rad dose) as summarized in Table V-22 (BEIR III, p. 209). As noted above, this estimate can probably be considered as the most plausible of the estimates given. It is emphasized that although the linear-quadratic model provides for reduced effects (per rad) for exposures

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(a) BEIR III also provides estimates of the effects of exposures of one rad per year either persisting throughout a lifetime or for various ages intended to reflect occupational exposures. These alternative estimates have not been used, because they are based on the assumption that all persons at risk receive exposure from birth to the end of life (or from particular specified ages in the case of occupational exposure). In the situation of interest in this report, it is expected that persons will begin exposure at different ages, and that as persons move away or leave their jobs, others will replace them. With the linear-quadratic function used in BEIR III, the effect (per rad) of 10 rad is not that much different than the effect of one rad. The slight per-rad increase resulting from using the 10-rad situation as opposed to using the single-rad exposure situation will be the same for either situation (e.g., capture vs. release) being compared in this study.

(b) Although the magnitude of the risks predicted by other BEIR III models would differ from those given in this report, the ratios of the estimates for the three exposure situations examined here would be the same, and thus the comparison of these estimates is unaffected by the choice between linear and linear-quadratic models and between absolute and relative risk models.

TABLE 6.1. Comparison of Estimated Cancer Mortality from the Capture and Immobilization of ^{85}Kr as Opposed to Routine Release to the Environment

Exposure Situation	Lifetime Risk/Person per Year Exposure[(a)]	Total Number of Radiation-Induced Cancer Deaths Expected from 30 Years of Exposure
Occupational (Capture)		
450 mrem/yr 9 people	1.0×10^{-4}	0.027
570 mrem/yr 7 people	1.3×10^{-4}	0.027
425 mrem/yr 7 people	9.6×10^{-5}	0.020
600 mrem/yr 5 people	1.4×10^{-4}	0.020
General Public (Routine Release)		
0.003 mrem/yr 850,000 people	6.8×10^{-10}	0.017

(a) 2255 deaths/10^6 people/yr for 10 rad dose

at ten rad per year as opposed to exposure of 100 rads or more, it does not provide for reduced effects from the dose rates involved in ^{85}Kr release to the environment as opposed to the higher dose rates involved in occupational exposures.

It is emphasized that these estimates are based on the assumption that at the levels being compared, risks are proportional to dose. Data from animal experiments indicate that exposures at low dose rates (in the range of 10 rad per day) convey less risk (per rad) than do exposures at high dose rates (in the range of 50 rad per minute). Unfortunately, there are no adequate data to allow comparing effects of the 500-mrem/year occupational dose to the 0.003-mrem/year population dose developed in this study. It is possible that no risks would occur from either situation (e.g., capture vs. release).

Given the uncertainties in the models used to generate these estimates, the differences in total risks for the situations considered here cannot be considered meaningful. Certainly there is no compelling reason to conclude that risks from ^{85}Kr release are greater than those from capturing and immobilizing ^{85}Kr.

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APPENDIX A

SOURCE-TERM SENSITIVITY ANALYSES

APPENDIX A

SENSITIVITY ANALYSIS OF ^{85}Kr RELEASES FROM AGED FUEL AS A FUNCTION OF THREE REPROCESSING SCENARIOS

Most environmental assessments involving nuclear fuel reprocessing assume that the LWR fuel is about 160 or 180 days old at separation. However, the U.S. will probably not have any operational fuel reprocessing plants (FRP) before the year 2000. By that time there will be about 60,000 t of spent fuel in storage. It is assumed here that oldest fuel will be reprocessed first. Therefore a U.S. FRP is not likely to reprocess fuel less than 30 years old.

Three reprocessing scenarios have been evaluated to 1) provide a more realistic estimate of the ^{85}Kr releases from an FRP separating aged fuel, and 2) explore the needed FRP capacity to separate the backlog of spent fuel within a reasonable time frame.

The decay algorithm, used to estimate the ^{85}Kr source term from decayed fuel, requires a decay time as input which is determined from the spent fuel data by subtracting the year that the spent fuel was produced from the year that the spent fuel is expected to be reprocessed. The decay algorithm uses only one decay time for each year of reprocessing and not for each year the fuel was produced, even though several years worth of fuel production may be reprocessed in the same year. If more than one year's production of spent fuel is reprocessed in one year, then the decay time of the newest spent fuel is used by the decay algorithm. This is conservative since some of the fuel reprocessed under this scenario has a longer decay time and, therefore, less radioactivity. Under this scenario, the decay algorithm actually over-predicts the ^{85}Kr inventory.

The initial ^{85}Kr release from 1500 t of reprocessed fuel is conservatively estimated to be about 3×10^6 Ci as shown in Table A.1. This value is a reduction of about 5 the number of curies of ^{85}Kr contained in 180-day-cooled fuel.

TABLE A.1. Reprocessing Scenario A

END OF YEAR	FUEL SUPPLIED (MTIHM)		FUEL PROCESSED (MTIHM)		NET FUEL (MTIHM)	YEARLY KR85 DISCHARGED (CI)	ACCUM. KR85 DISCHARGED (CI)
	ANNUAL	ACCUM	ANNUAL	ACCUM			
1971	124	124	0	0	124	0.00E+00	0.00E+00
1972	291	415	0	0	415	0.00E+00	0.00E+00
1973	165	580	0	0	580	0.00E+00	0.00E+00
1974	447	1027	0	0	1027	0.00E+00	0.00E+00
1975	563	1590	0	0	1590	0.00E+00	0.00E+00
1976	682	2272	0	0	2272	0.00E+00	0.00E+00
1977	860	3132	0	0	3132	0.00E+00	0.00E+00
1978	1150	4282	0	0	4282	0.00E+00	0.00E+00
1979	1204	5486	0	0	5486	0.00E+00	0.00E+00
1980	1149	6635	0	0	6635	0.00E+00	0.00E+00
1981	1265	7900	0	0	7900	0.00E+00	0.00E+00
1982	1089	8989	0	0	8989	0.00E+00	0.00E+00
1983	1300	10289	0	0	10289	0.00E+00	0.00E+00
1984	1618	11907	0	0	11907	0.00E+00	0.00E+00
1985	1905	13812	0	0	13812	0.00E+00	0.00E+00
1986	2207	16019	0	0	16019	0.00E+00	0.00E+00
1987	2493	18512	0	0	18512	0.00E+00	0.00E+00
1988	2752	21264	0	0	21264	0.00E+00	0.00E+00
1989	2572	23836	0	0	23836	0.00E+00	0.00E+00
1990	3238	27074	0	0	27074	0.00E+00	0.00E+00
1991	2931	30005	0	0	30005	0.00E+00	0.00E+00
1992	2886	32891	0	0	32891	0.00E+00	0.00E+00
1993	3123	36014	0	0	36014	0.00E+00	0.00E+00
1994	2993	39007	0	0	39007	0.00E+00	0.00E+00
1995	2995	42002	0	0	42002	0.00E+00	0.00E+00
1996	3208	45210	0	0	45210	0.00E+00	0.00E+00
1997	3010	48220	0	0	48220	0.00E+00	0.00E+00
1998	3090	51310	0	0	51310	0.00E+00	0.00E+00
1999	3467	54777	0	0	54777	0.00E+00	0.00E+00
2000	3110	57887	0	0	57887	0.00E+00	0.00E+00
2001	3027	60914	750	750	60164	1.34E+06	1.34E+06
2002	3618	64532	750	1500	63032	1.34E+06	2.68E+06
2003	3411	67943	750	2250	65693	1.34E+06	4.03E+06
2004	3623	71566	750	3000	68566	1.34E+06	5.37E+06
2005	3888	75454	750	3750	71704	1.34E+06	6.71E+06
2006	4132	79586	1500	5250	74336	2.68E+06	9.39E+06
2007	4429	84015	1500	6750	77265	2.87E+06	1.23E+07
2008	5176	89191	1500	8250	80941	2.87E+06	1.51E+07
2009	4391	93582	1500	9750	83832	2.87E+06	1.80E+07
2010	4497	98079	1500	11250	86829	2.87E+06	2.09E+07
2011	4639	102718	2250	13500	89218	4.30E+06	2.52E+07
2012	4625	107343	2250	15750	91593	4.30E+06	2.95E+07
2013	4773	112116	2250	18000	94116	4.30E+06	3.37E+07
2014	4948	117064	2250	20250	96814	4.30E+06	3.80E+07
2015	5086	122150	2250	22500	99650	4.30E+06	4.23E+07
2016	4964	127114	2250	24750	102364	4.30E+06	4.66E+07
2017	5272	132386	2250	27000	105386	4.03E+06	5.07E+07
2018	5215	137601	2250	29250	108351	4.03E+06	5.47E+07
2019	5024	142625	2250	31500	111125	4.03E+06	5.87E+07
2020	5194	147819	2250	33750	114069	4.03E+06	6.27E+07

END OF YEAR	FUEL SUPPLIED (MTIHM)		FUEL PROCESSED (MTIHM)		NET FUEL (MTIHM)	YEARLY KR85 DISCHARGED (CI)	ACCUM. KR85 DISCHARGED (CI)
	ANNUAL	ACCUM	ANNUAL	ACCUM			
2021	0	147819	2250	36000	111819	3.77E+06	6.65E+07
2022	0	147819	2250	38250	109569	3.77E+06	7.03E+07
2023	0	147819	2250	40500	107319	3.77E+06	7.41E+07
2024	0	147819	2250	42750	105069	3.77E+06	7.78E+07
2025	0	147819	2250	45000	102819	3.53E+06	8.14E+07
2026	0	147819	2250	47250	100569	3.53E+06	8.49E+07
2027	0	147819	2250	49500	98319	3.53E+06	8.84E+07
2028	0	147819	2250	51750	96069	3.53E+06	9.20E+07
2029	0	147819	2250	54000	93819	3.31E+06	9.53E+07
2030	0	147819	2250	56250	91569	3.31E+06	9.86E+07
2031	0	147819	2250	58500	89319	3.31E+06	1.02E+08
2032	0	147819	2250	60750	87069	3.10E+06	1.05E+08
2033	0	147819	2250	63000	84819	3.10E+06	1.08E+08
2034	0	147819	2250	65250	82569	3.10E+06	1.11E+08
2035	0	147819	2250	67500	80319	2.90E+06	1.14E+08
2036	0	147819	2250	69750	78069	2.90E+06	1.17E+08
2037	0	147819	2250	72000	75819	2.90E+06	1.20E+08
2038	0	147819	2250	74250	73569	2.72E+06	1.23E+08
2039	0	147819	2250	76500	71319	2.72E+06	1.25E+08
2040	0	147819	2250	78750	69069	2.54E+06	1.28E+08
2041	0	147819	2250	81000	66819	2.54E+06	1.30E+08
2042	0	147819	2250	83250	64569	2.38E+06	1.33E+08
2043	0	147819	2250	85500	62319	2.38E+06	1.35E+08
2044	0	147819	2250	87750	60069	2.23E+06	1.37E+08
2045	0	147819	2250	90000	57819	2.23E+06	1.40E+08
2046	0	147819	2250	92250	55569	2.09E+06	1.42E+08
2047	0	147819	2250	94500	53319	2.09E+06	1.44E+08
2048	0	147819	2250	96750	51069	1.96E+06	1.46E+08
2049	0	147819	2250	99000	48819	1.96E+06	1.48E+08
2050	0	147819	2250	101250	46569	1.83E+06	1.50E+08
2051	0	147819	2250	103500	44319	1.83E+06	1.51E+08
2052	0	147819	2250	105750	42069	1.72E+06	1.53E+08
2053	0	147819	2250	108000	39819	1.72E+06	1.55E+08
2054	0	147819	2250	110250	37569	1.61E+06	1.56E+08
2055	0	147819	2250	112500	35319	1.61E+06	1.58E+08
2056	0	147819	2250	114750	33069	1.51E+06	1.60E+08
2057	0	147819	2250	117000	30819	1.41E+06	1.61E+08
2058	0	147819	2250	119250	28569	1.41E+06	1.62E+08
2059	0	147819	2250	121500	26319	1.32E+06	1.64E+08
2060	0	147819	2250	123750	24069	1.32E+06	1.65E+08
2061	0	147819	2250	126000	21819	1.24E+06	1.66E+08
2062	0	147819	2250	128250	19569	1.24E+06	1.67E+08
2063	0	147819	2250	130500	17319	1.16E+06	1.69E+08
2064	0	147819	2250	132750	15069	1.16E+06	1.70E+08
2065	0	147819	2250	135000	12819	1.09E+06	1.71E+08
2066	0	147819	2250	137250	10569	1.02E+06	1.72E+08
2067	0	147819	2250	139500	8319	1.02E+06	1.73E+08
2068	0	147819	2250	141750	6069	9.52E+05	1.74E+08
2069	0	147819	2250	144000	3819	9.52E+05	1.75E+08
2070	0	147819	2250	146250	1569	8.92E+05	1.76E+08

Three reprocessing scenarios were generated to examine how long the U.S. would have to maintain a viable reprocessing industry to separate the backlog of spent fuel. Scenario (case) A consists of three 750 tHM/yr FRPs coming on line five years apart beginning with the first one in 2001. Case B projects the impacts upon the stored fuel backlog with two 1500 tHM/yr FRPs coming on line in 2001 and 2006, respectively. The third case (C) extends case B to a third 1500 tHM plant due in 2011.

The potential impacts upon the volume of stored spent fuel attributed to these scenarios is shown in tabular form in Tables A.1, A.2, and A.3 for cases A, B and C, respectively, and summarized here and in Figure A.1.

Under the assumption that no additional spent fuel is generated after the year 2020, the backlog would be reprocessed under case A by the year 2072; case B, by the year 2053, and case C, by the year 2039. Case C (three 1500 tHM/yr FRPs) would complete the reprocessing of backlogged fuel 33 years sooner than case A and 14 years sooner than case B.

TABLE A.2. Reprocessing Scenario B

END OF YEAR	FUEL SUPPLIED (MTIHM)		FUEL PROCESSED (MTIHM)		NET FUEL (MTIHM)	YEARLY KR85 DISCHARGED (CI)	ACCUM. KR85 DISCHARGED (CI)
	ANNUAL	ACCUM	ANNUAL	ACCUM			
1971	124	124	0	0	124	0.00E+00	0.00E+00
1972	291	415	0	0	415	0.00E+00	0.00E+00
1973	165	580	0	0	580	0.00E+00	0.00E+00
1974	447	1027	0	0	1027	0.00E+00	0.00E+00
1975	563	1590	0	0	1590	0.00E+00	0.00E+00
1976	682	2272	0	0	2272	0.00E+00	0.00E+00
1977	860	3132	0	0	3132	0.00E+00	0.00E+00
1978	1150	4282	0	0	4282	0.00E+00	0.00E+00
1979	1204	5486	0	0	5486	0.00E+00	0.00E+00
1980	1149	6635	0	0	6635	0.00E+00	0.00E+00
1981	1265	7900	0	0	7900	0.00E+00	0.00E+00
1982	1089	8989	0	0	8989	0.00E+00	0.00E+00
1983	1300	10289	0	0	10289	0.00E+00	0.00E+00
1984	1618	11907	0	0	11907	0.00E+00	0.00E+00
1985	1905	13812	0	0	13812	0.00E+00	0.00E+00
1986	2207	16019	0	0	16019	0.00E+00	0.00E+00
1987	2493	18512	0	0	18512	0.00E+00	0.00E+00
1988	2752	21264	0	0	21264	0.00E+00	0.00E+00
1989	2572	23836	0	0	23836	0.00E+00	0.00E+00
1990	3238	27074	0	0	27074	0.00E+00	0.00E+00
1991	2931	30005	0	0	30005	0.00E+00	0.00E+00
1992	2886	32891	0	0	32891	0.00E+00	0.00E+00
1993	3123	36014	0	0	36014	0.00E+00	0.00E+00
1994	2993	39007	0	0	39007	0.00E+00	0.00E+00
1995	2995	42002	0	0	42002	0.00E+00	0.00E+00
1996	3208	45210	0	0	45210	0.00E+00	0.00E+00
1997	3010	48220	0	0	48220	0.00E+00	0.00E+00
1998	3090	51310	0	0	51310	0.00E+00	0.00E+00
1999	3467	54777	0	0	54777	0.00E+00	0.00E+00
2000	3110	57887	0	0	57887	0.00E+00	0.00E+00
2001	3027	60914	1500	1500	59414	2.87E+06	2.87E+06
2002	3618	64532	1500	3000	61532	3.06E+06	5.93E+06
2003	3411	67943	1500	4500	63443	3.27E+06	9.19E+06
2004	3623	71566	1500	6000	65566	3.27E+06	1.25E+07
2005	3888	75454	1500	7500	67954	3.27E+06	1.57E+07
2006	4132	79586	3000	10500	69086	7.45E+06	2.32E+07
2007	4429	84015	3000	13500	70515	7.45E+06	3.06E+07
2008	5176	89191	3000	16500	72691	7.95E+06	3.86E+07
2009	4391	93582	3000	19500	74082	7.95E+06	4.65E+07
2010	4497	98079	3000	22500	75579	7.95E+06	5.45E+07
2011	4639	102718	3000	25500	77218	7.95E+06	6.24E+07
2012	4625	107343	3000	28500	78843	7.95E+06	7.04E+07
2013	4773	112116	3000	31500	80616	7.95E+06	7.83E+07
2014	4948	117064	3000	34500	82564	7.95E+06	8.63E+07
2015	5086	122150	3000	37500	84650	7.95E+06	9.43E+07
2016	4964	127114	3000	40500	86614	7.95E+06	1.02E+08
2017	5272	132386	3000	43500	88886	7.95E+06	1.10E+08
2018	5215	137601	3000	46500	91101	7.95E+06	1.18E+08
2019	5024	142625	3000	49500	93125	7.95E+06	1.26E+08
2020	5194	147819	3000	52500	95319	7.95E+06	1.34E+08

END OF YEAR	FUEL SUPPLIED (MTIHM)		FUEL PROCESSED (MTIHM)		NET FUEL (MTIHM)	YEARLY KR85 DISCHARGED (CI)	ACCUM. KR85 DISCHARGED (CI)
	ANNUAL	ACCUM	ANNUAL	ACCUM			
2021	0	147819	3000	55500	92319	7.95E+06	1.42E+08
2022	0	147819	3000	58500	89319	7.95E+06	1.50E+08
2023	0	147819	3000	61500	86319	7.95E+06	1.58E+08
2024	0	147819	3000	64500	83319	7.45E+06	1.65E+08
2025	0	147819	3000	67500	80319	7.45E+06	1.73E+08
2026	0	147819	3000	70500	77319	7.45E+06	1.80E+08
2027	0	147819	3000	73500	74319	7.45E+06	1.88E+08
2028	0	147819	3000	76500	71319	7.45E+06	1.95E+08
2029	0	147819	3000	79500	68319	6.98E+06	2.02E+08
2030	0	147819	3000	82500	65319	6.98E+06	2.09E+08
2031	0	147819	3000	85500	62319	6.98E+06	2.16E+08
2032	0	147819	3000	88500	59319	6.53E+06	2.23E+08
2033	0	147819	3000	91500	56319	6.53E+06	2.29E+08
2034	0	147819	3000	94500	53319	6.53E+06	2.36E+08
2035	0	147819	3000	97500	50319	6.12E+06	2.42E+08
2036	0	147819	3000	100500	47319	6.12E+06	2.48E+08
2037	0	147819	3000	103500	44319	6.12E+06	2.54E+08
2038	0	147819	3000	106500	41319	5.73E+06	2.60E+08
2039	0	147819	3000	109500	38319	5.73E+06	2.65E+08
2040	0	147819	3000	112500	35319	5.73E+06	2.71E+08
2041	0	147819	3000	115500	32319	5.37E+06	2.77E+08
2042	0	147819	3000	118500	29319	5.37E+06	2.82E+08
2043	0	147819	3000	121500	26319	5.03E+06	2.87E+08
2044	0	147819	3000	124500	23319	5.03E+06	2.92E+08
2045	0	147819	3000	127500	20319	5.03E+06	2.97E+08
2046	0	147819	3000	130500	17319	4.71E+06	3.02E+08
2047	0	147819	3000	133500	14319	4.71E+06	3.06E+08
2048	0	147819	3000	136500	11319	4.41E+06	3.11E+08
2049	0	147819	3000	139500	8319	4.41E+06	3.15E+08
2050	0	147819	3000	142500	5319	4.13E+06	3.19E+08
2051	0	147819	3000	145500	2319	4.13E+06	3.24E+08
2052	0	147819	3000	148500	-681	4.13E+06	3.28E+08

TABLE A.3. Reprocessing Scenario C

END OF YEAR	FUEL SUPPLIED (MTIHM)		FUEL PROCESSED (MTIHM)		NET FUEL (MTIHM)	YEARLY KR85 DISCHARGED (CI)	ACCUM. KR85 DISCHARGED (CI)
	ANNUAL	ACCUM	ANNUAL	ACCUM			
1971	124	124	0	0	124	0.00E+00	0.00E+00
1972	291	415	0	0	415	0.00E+00	0.00E+00
1973	165	580	0	0	580	0.00E+00	0.00E+00
1974	447	1027	0	0	1027	0.00E+00	0.00E+00
1975	563	1590	0	0	1590	0.00E+00	0.00E+00
1976	682	2272	0	0	2272	0.00E+00	0.00E+00
1977	860	3132	0	0	3132	0.00E+00	0.00E+00
1978	1150	4282	0	0	4282	0.00E+00	0.00E+00
1979	1204	5486	0	0	5486	0.00E+00	0.00E+00
1980	1149	6635	0	0	6635	0.00E+00	0.00E+00
1981	1265	7900	0	0	7900	0.00E+00	0.00E+00
1982	1089	8989	0	0	8989	0.00E+00	0.00E+00
1983	1300	10289	0	0	10289	0.00E+00	0.00E+00
1984	1618	11907	0	0	11907	0.00E+00	0.00E+00
1985	1905	13812	0	0	13812	0.00E+00	0.00E+00
1986	2207	16019	0	0	16019	0.00E+00	0.00E+00
1987	2493	18512	0	0	18512	0.00E+00	0.00E+00
1988	2752	21264	0	0	21264	0.00E+00	0.00E+00
1989	2572	23836	0	0	23836	0.00E+00	0.00E+00
1990	3238	27074	0	0	27074	0.00E+00	0.00E+00
1991	2931	30005	0	0	30005	0.00E+00	0.00E+00
1992	2886	32891	0	0	32891	0.00E+00	0.00E+00
1993	3123	36014	0	0	36014	0.00E+00	0.00E+00
1994	2993	39007	0	0	39007	0.00E+00	0.00E+00
1995	2995	42002	0	0	42002	0.00E+00	0.00E+00
1996	3208	45210	0	0	45210	0.00E+00	0.00E+00
1997	3010	48220	0	0	48220	0.00E+00	0.00E+00
1998	3090	51310	0	0	51310	0.00E+00	0.00E+00
1999	3467	54777	0	0	54777	0.00E+00	0.00E+00
2000	3110	57887	0	0	57887	0.00E+00	0.00E+00
2001	3027	60914	1500	1500	59414	2.87E+06	2.87E+06
2002	3618	64532	1500	3000	61532	3.06E+06	5.93E+06
2003	3411	67943	1500	4500	63443	3.27E+06	9.19E+06
2004	3623	71566	1500	6000	65566	3.27E+06	1.25E+07
2005	3888	75454	1500	7500	67954	3.27E+06	1.57E+07
2006	4132	79586	3000	10500	69086	7.45E+06	2.32E+07
2007	4429	84015	3000	13500	70515	7.45E+06	3.06E+07
2008	5176	89191	3000	16500	72691	7.95E+06	3.86E+07
2009	4391	93582	3000	19500	74082	7.95E+06	4.65E+07
2010	4497	98079	3000	22500	75579	7.95E+06	5.45E+07
2011	4639	102718	4500	27000	75718	1.19E+07	6.64E+07
2012	4625	107343	4500	31500	75843	1.27E+07	7.92E+07
2013	4773	112116	4500	36000	76116	1.27E+07	9.19E+07
2014	4948	117064	4500	40500	76564	1.36E+07	1.05E+08
2015	5086	122150	4500	45000	77150	1.36E+07	1.19E+08
2016	4964	127114	4500	49500	77614	1.45E+07	1.34E+08
2017	5272	132386	4500	54000	78386	1.45E+07	1.48E+08
2018	5215	137601	4500	58500	79101	1.55E+07	1.64E+08
2019	5024	142625	4500	63000	79625	1.55E+07	1.79E+08
2020	5194	147819	4500	67500	80319	1.55E+07	1.95E+08

END OF YEAR	FUEL SUPPLIED (MTIHM) ANNUAL	ACCUM	FUEL PROCESSED (MTIHM) ANNUAL	ACCUM	NET FUEL (MTIHM)	YEARLY KR85 DISCHARGED (CI)	ACCUM. KR85 DISCHARGED (CI)
2021	0	147819	4500	72000	75819	1.66E+07	2.11E+08
2022	0	147819	4500	76500	71319	1.66E+07	2.28E+08
2023	0	147819	4500	81000	66819	1.66E+07	2.44E+08
2024	0	147819	4500	85500	62319	1.66E+07	2.61E+08
2025	0	147819	4500	90000	57819	1.66E+07	2.77E+08
2026	0	147819	4500	94500	53319	1.66E+07	2.94E+08
2027	0	147819	4500	99000	48819	1.66E+07	3.11E+08
2028	0	147819	4500	103500	44319	1.66E+07	3.27E+08
2029	0	147819	4500	108000	39819	1.66E+07	3.44E+08
2030	0	147819	4500	112500	35319	1.66E+07	3.60E+08
2031	0	147819	4500	117000	30819	1.55E+07	3.76E+08
2032	0	147819	4500	121500	26319	1.55E+07	3.91E+08
2033	0	147819	4500	126000	21819	1.55E+07	4.07E+08
2034	0	147819	4500	130500	17319	1.55E+07	4.22E+08
2035	0	147819	4500	135000	12819	1.55E+07	4.38E+08
2036	0	147819	4500	139500	8319	1.55E+07	4.53E+08
2037	0	147819	4500	144000	3819	1.55E+07	4.69E+08
2038	0	147819	4500	148500	-681	1.55E+07	4.84E+08

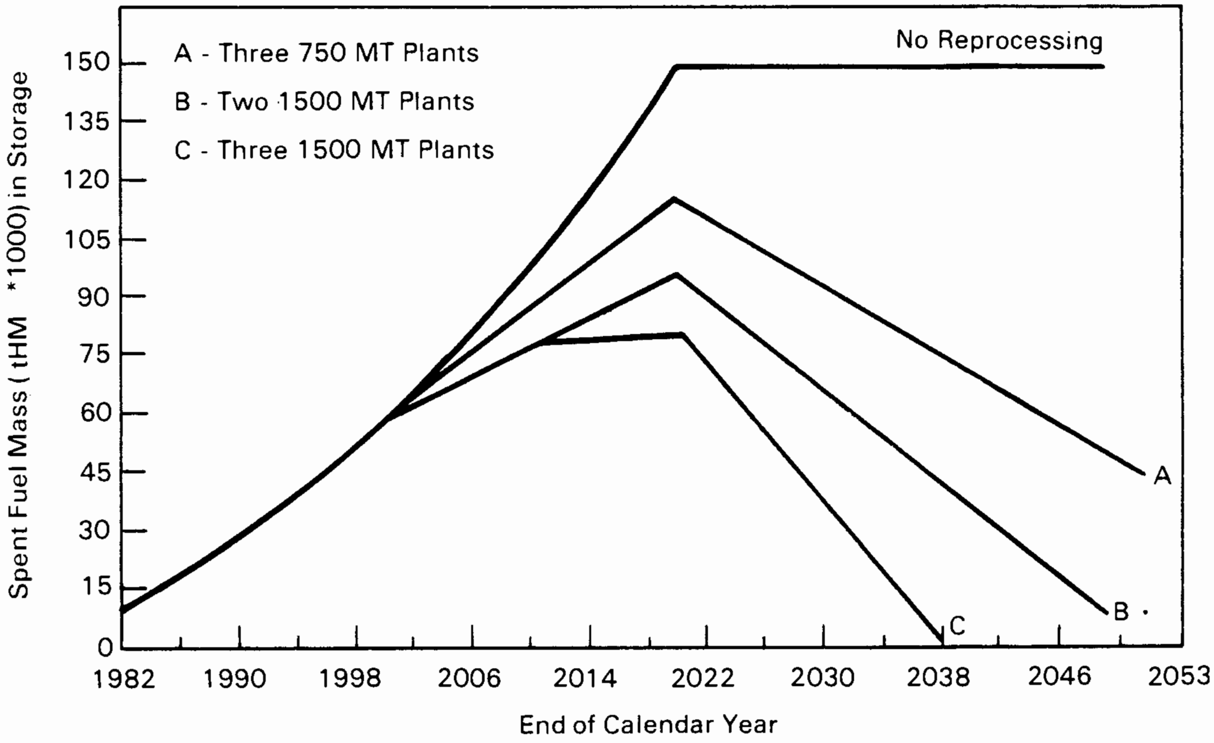

FIGURE A.1. Spent Fuel in Storage with Three Reprocessing Scenarios

APPENDIX B

METHODS FOR CALCULATING OCCUPATIONAL DOSES FROM KRYPTON-85 EXPOSURE

APPENDIX B

METHODS FOR CALCULATING OCCUPATIONAL DOSES FROM ^{85}Kr EXPOSURE

B.1 ISOSHLD-II

ISOSHLD-II [an extension of ISOSHLD (Engle et al. 1966) both developed at Pacific Northwest Laboratory] is a computer code used to perform beta- and gamma-ray shielding calculations for isotopic sources in a variety of source and shield configurations. The energy distribution of bremsstrahlung produced by the slowing down and stopping of beta particles is assumed to be produced only in the source region. Attenuation calculations are performed by point kernel integration; for most geometries by Simpson's rule numerical integration. Buildup factors are calculated by the code based on the number of mean-free paths of material between the source and dose points, the effective atomic number of a particular shield region, and the point isotropic buildup data available as Taylor coefficients. Other data needed to solve most isotope shielding of practical interest are linked to ISOSHLD in various libraries. For most problems the user need only supply:

- The geometry and material composition of the source
- The geometry and materials of the shields.

B.2 CYLIN

CYLIN is a point kernel shielding code developed at Pacific Northwest Laboratory as a modification of the shielding code PERCS (Reece et al. 1984). The modified code computes doses from gamma emitting radioisotopes within a cylindrical volume, whereas PERCS computes the doses from sources located on the interior surfaces of cylinders. The self-shielding of the source is also taken into account.

The evaluation of the point kernel equation for a cylindrical source is a triple integral of the form

$$\text{Dose} = \frac{1}{4\pi} \int_{\text{ALOW}}^{\text{AHIGH}} \int_{0}^{R} \int_{0}^{2\pi} \frac{\text{SA*[BLDUP*E*YIELD*K(E)*EXP(-BSUM)] r dALPHA dA dr}}{\text{RHO}^2}$$

where AHIGH and ALOW = the top and bottom elevation of the source

R = the inside radius of the source;

RHO = the distance between the differential source point and the dose point;

SA = the differential source strength, μCi/cm^3;

BLDUP = the ratio of the dose from the total flux to the dose from the uncollided flux;

E = the energy of the particular γ-ray emitted during the isotope decay process, MeV;

YIELD = the probability of a particular γ-ray being emitted during the decay process;

K(E) = a factor which converts MeV/cm^2/sec to mrem/hr for a given γ-ray energy;

BSUM = the sum of all the optical thicknesses along a source-dose line;

r = the running variable of integration along the inside radius of the source;

alpha = the running variable of integration about all angles;

a = the running variable of integration of length along the source.

For every dose point CYLIN evaluates the contribution of each source volume by calculating all the parameters such as upper and lower limits of integration and initiating the numerical integration of the point kernel. The triple integration through the angle, radius, and length of the cylindrical volume is done by spline quadrature routines or Newton-Cotes quadrature routines, depending on the location of the dose point relative to the source. As each differential source element is chosen, all the cylinders and auxiliary shields are checked to determine if they lie between the source element and the dose point. A tally is kept of the contribution to optical thickness and the relative location of each shield.

After the geometric calculations are done, the concentration of each isotope in the differential volume is computed and each gamma energy and yield for that isotope is used, along with the calculated buildup factor, to evaluate the point kernel equation. The quadrature routines are then used to calculate the integral. This is done for each source volume element and the total dose is the sum of the contribution from each source. The next dose point is then prompted from the user and the calculations are repeated.

The program CYLIN calculates the dose from up to twenty cylindrical sources and cylindrical or rectangular shields in any configuration. PERCS has been tested against other current shielding codes with good results and CYLIN dose values are compared in the following table against the well-known point kernel shielding code ISOSHLD-II.

Cylinder Filled With ^{85}Kr

Dose Points	CYLIN	ISOSHLD-II
4' elevation at surface	25.65 mrem/hr	27.17 mrem/hr
12' elevation at surface	25.98 mrem/hr	27.20 mrem/hr
4' elevation at 3' away	2.702 mrem/hr	2.622 mrem/hr
12' elevation at 3' away	2.936 mrem/hr	2.951 mrem/hr

cylinder: radius = 11.43 cm; height = 731.5 cm; filled with 10 curies ^{85}Kr; wall thickness = 0.93 cm; wall material-steel; no shielding.

Some of the principal advantages of CYLIN are as follows:

1. Completely arbitrary geometry is allowed in cylinder location, size, and source content (both spatially and isotopically).
2. The location of the dose point is completely arbitrary, including inside a cylinder or shield. The single exception is the placement of the dose point exactly on the centerline of the cylindrical source.
3. The shielding of the distributed source by the cylinder and its contents is calculated.
4. All other cylindrical sources and shields on a line-of-sight between the differential source point and the dose point are included in the buildup calculation.

APPENDIX C

EQUATIONS FOR CALCULATING DOSES FROM ^{85}KR EXPOSURE TO THE PUBLIC

APPENDIX C

EQUATIONS FOR CALCULATING DOSES FROM ^{85}Kr EXPOSURE TO THE PUBLIC

Several equations are used to characterize the segment of the population being exposed and the conditions of the ^{85}Kr release. The population segments considered are the maximally exposed individual and the regional population.

C.1 MAXIMUM INDIVIDUAL DOSE

The maximum individual dose assumes that an individual continuously resides at the point of highest ground level concentration of ^{85}Kr released from the BNFP stack. Depending upon the release height, terrain and meteorology, this point is generally within a few miles of the plant. Equation 1 is the annual dose equation multiplied by 30 years to obtain a 30-year cumulative dose.

$$\mathrm{Dose}_{30} = \frac{(\mathrm{Ci/yr})(\bar{X}/Q' \ \mathrm{sec/m^3})(10^{12}\ \mathrm{pCi/Ci})}{3.15 \times 10^{7}\ \mathrm{sec/yr}} \, (\mathrm{Df\ rem/yr\ per\ pCi/m^3})(30\ \mathrm{yr}) \qquad (1)$$

where Ci/yr is the annual ^{85}Kr effluent;

$\bar{X}/Q'$ is the annual average atmospheric dispersion factor, sec/m^3;

Df is the external dose factor, 1.9E-8 rem/yr per pCi/m^3;

30 is the operational life time of the FRP.

The annual average atmospheric dispersion factors ($\bar{X}/Q'$) calculated for the BNFP Separations Facility main stack of 100m height are presented in Table 4.2. By inspection of the data, the maximum offsite ground-level concentration ($\bar{X}/Q'$ = 1.3E-8 sec/m^3) occurs 1 to 2 miles from the plant in the E direction.

The external dose factors (Df) used here are calculated as shown by Soldat et al. (1976). However, for comparison, Table C.1 lists published

TABLE C.1. Comparison of Several External Dose Factors for ^{85}Kr in the Literature (rem/yr per pCi/m^3)

β-Skin	γ-Skin	β+γ-Skin	γ-Testes	γ-Total Body	Reference
1.4E-6	2.4E-8	1.4E-6	1.9E-8	1.9E-8	Soldat et al. 1976
1.3E-6	---	---	---	1.6E-8	NRC 1977
1.5E-6	---	1.5E-6	2.9E-8	2.9E-8	Schaeffer 1973
---	---	1.5E-6	---	1.6E-8	Strenge 1981
2.2E-6	1.8E-8	2.2E-6	---	1.8E-8	Killough et al. 1976

values by several workers. The assumptions used by these authors were not always the same. This is especially true of the depth at which the "skin" dose was calculated. The dose to the internal organs, including the lung, from external exposure is generally assumed to be the same as the total-body dose. To estimate the lung dose from inhalation, Soldat et al. (1976) calculated dose factors from the ICRP initial lung model (ICRP 1959). The concentration of the noble gas in the air within the lung is assumed to be the same as that in the inspired air. The effective energies of the individual radiations are calculated using an effective radius of 10 cm for the lung.

The inhalation dose factor calculated by Soldat is compared in Table C.2 with values obtained by other researchers. Caution should be used when making a critical comparison of these values. For example, Russell and Galpin (1972) used a lung volume of 5.6 liters, Soldat et al. (1976) based their calculations on a lung volume of 4 liters, and Whitton (1968) did not state the lung volume used for her calculations. When the variations in assumptions are factored into the calculation of the dose factors by these various researchers, there is reasonable agreement among the values.

The total dose would of course be the sum of the external component and the internal contribution. The doses from ^{85}Kr absorbed in tissue is generally small compared to the dose from direct external radiation. It is apparent that for ^{85}Kr the skin is the critical organ for individuals. For collective dose calculations the generally accepted practice is to calculate total-body doses so that health effects could then be estimated if desired.

C.2 50-MILE POPULATION DOSE

The dose to the 50-mile population is the annual dose multiplied by 30 years (Equation 2).

$$\mathrm{Dose}_{30} = \frac{(\mathrm{Ci/yr}) \sum (\bar{\chi}/Q')(\mathrm{POP}) \frac{\mathrm{sec} \cdot \mathrm{man}}{\mathrm{m}^3}}{3.15 \times 10^7 \ \mathrm{sec/yr}} (10^{12} \ \mathrm{pCi/Ci})\left(\mathrm{Df} \ \frac{\mathrm{rem}}{\mathrm{yr}} \ \mathrm{per} \ \frac{\mathrm{pCi}}{\mathrm{m}^3}\right)(30 \ \mathrm{yr}) \tag{2}$$

where: $[(\bar{\chi}/Q')(\mathrm{POP})]$ is the integrated population times annual average atmospheric dispersion factor, sec $\cdot$ man/m^3.

TABLE C.2. Comparison of Inhalation Dose Factors for ^{85}Kr and the Lung (rem/yr per pCi/m^3)

DCF (Lung)	Reference
1.7E-8	Soldat et al. 1976
5.1E-8	Schaeffer 1973
1.4E-8	Whitton 1968
2.6E-8	Russell and Galpin 1972
1.8E-8	Snyder et al. 1975

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