W. R. GRACE & CO.
RESEARCH DIVISION
Washington Research Center, Clarksville, Maryland 21029

GRACE

70-456
For Div. of Compliance

APPLICATION FOR AMENDMENT TO SPECIAL NUCLEAR MATERIALS LICENSE SNM-840 FOR NUCLEAR CHEMISTRY FACILITY

RECEIVED
APR 9 1970
U.S. ATOMIC ENERGY COMMISSION
Regulatory Mail Section

Copy No. 5

APRIL 8, 1970

C117
B/854

ITEM # 61

Cable - Gracerad   Tel. (301) 531 - 5711   TWX - 710 826 1886

(122)

DOCKET NO. 70-456

TABLE OF CONTENTS

APPLICATION FOR AMENDMENT TO SPECIAL NUCLEAR MATERIALS LICENSE SNM-840 FOR NUCLEAR CHEMISTRY FACILITY

1.0 INTRODUCTION

1.1 The Research Division of W. R. Grace & Co. has been involved in nuclear fuel development for nearly a decade. This work has been carried out under source materials license SM-340 and under special nuclear materials license SNM-840. The latter license permits the possession of up to 1 kg of U-235 in enriched uranium.

1.2 It is now desired to expand the development operations involving special nuclear material, and the W. R. Grace & Co. hereby requests an amendment to SNM-840 to increase the quantity of uranium-235 in enriched uranium in its possession from 1 kilogram to 200 kilograms and to permit the handling and processing of that uranium as specified herein. This information is furnished in support of this application for amendment to SNM-840 as required by 10 CFR 70.

1.3 It is planned that special nuclear material will be received as a form of the oxide, but not as metal or hexafluoride. It will be converted to oxide particles of desired shapes and densities which may approach theoretical density. The uranium will generally be highly enriched in the U-235 isotope. Materials which cannot be reintroduced to the normal process will be shipped to other licensees for recovery of uranium values.

2.0 CORPORATE INFORMATION

2.1 Several changes have occurred in the principal corporate officers and in the business activities of W. R. Grace & Co. since the last filing on October 4, 1967. This section is revised herein to bring required corporate information up to date.

2.2 Corporate Officers and Ownership--W. R. Grace & Co. is a United States Corporation, organized under the laws of the state of Connecticut with principal offices at 7 Hanover Square, New York, New York. The Research Division has its principal offices and facilities on Route 32 in Howard County, Clarksville, Maryland. The voting stock of the company is publicly owned and is traded on the New York Stock Exchange. As of July 29, 1968, shares of outstanding preferred and common stock of W. R. Grace & Co. representing more than 95% of the ownership and voting power of the company were held of record by stockholders deemed to be U.S. citizens.

The President, Vice-Presidents, Secretary, Treasurer and all but two members of the Board of Directors are U.S. citizens. However, neither of these individuals has any contact or business with the Bldg. 16-A Nuclear Facility.

On the basis of the above information of record, W. R. Grace & Co. is not owned or controlled by an alien, foreign corporation or foreign government.

The Directors and principal officers of W. R. Grace & Co. are listed in the 1971 Annual Report which is incorporated as Appendix C.

2.3 Financial Qualification-- The financial qualifications of the company are hereby updated by incorporation of the W. R. Grace & Co. 1971 Annual Report--Appendix C.

3.0. LOCATION AND GENERAL DESCRIPTION OF WASHINGTON RESEARCH CENTER

3.1 Location--The Washington Research Center is located in a rural area in Howard County, Maryland. Clarksville is three miles to the west, Simpsonville is two miles to the east. Howard County is substantially rural, the principal urban area in the county is Columbia (with a population of 18,000) located four miles northeast of the Washington Research Center. The Washington Research Center employs approximately 500 people.

3.2 Site Description--The Center is located on a 147 acre tract of rolling farm and wooded land with northern most part of the acreage extending across a section of the Middle Patuxent River. The topographic map, Figure 3.2, shows the site and the building arrangement.

3.3 Description of Facilities --The facilities in which the increased quantities of special nuclear materials under this amendment, are to be stored and processes are located in authorized areas in Bldg. 16-A and Bldg. 20 shown on Figure 3.2. Small quantities, such as waste may be stored in steel drums for a period not to exceed 72 hours, in the enclosed yard area, outside of Bldg. 16-A.

3.4 The Medical Department, First Aid Room and Health and Safety Department are located in Bldg. 2 shown in Figure 3.2. The assembly area for the Bldg. 16 complex is located in the Lobby Conference Room of Bldg. 2, where the emergency cache is located. Emergency equipment is located in these various offices.

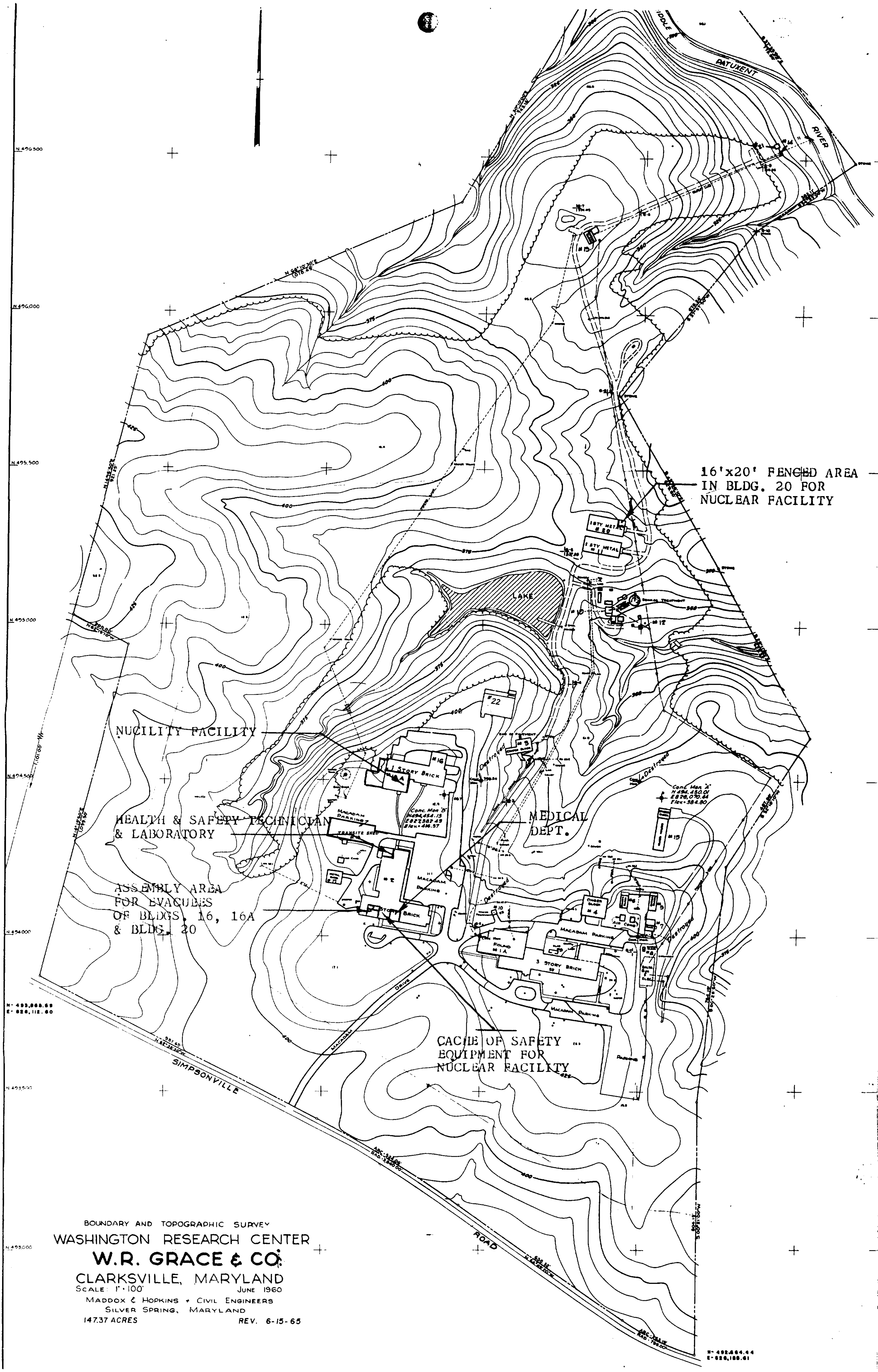
3.5 The normal SNM samples are analyzed in the analytical laboratories, see Figure 3.2. Special samples are run in supporting laboratories.

FIGURE 3.2

BOUNDARY AND TOPOGRAPHIC SURVEY

WASHINGTON RESEARCH CENTER

CLARKSVILLE, MARYLAND



16'x20' FENCED AREA IN BLDG. 20 FOR NUCLEAR FACILITY
NUCILITY FACILITY
HEALTH & SAFETY TECHNICIAN & LABORATORY
ASSEMBLY AREA FOR EVACUEES OF BLDGS. 16, 16A & BLDG. 20
MEDICAL DEPT.
CACHE OF SAFETY EQUIPMENT FOR NUCLEAR FACILITY
PATUXENT RIVER
LAKE
SIMPSONVILLE ROAD
MACADAM PARKING
3 STORY BRICK
BOUNDARY AND TOPOGRAPHIC SURVEY
WASHINGTON RESEARCH CENTER
W.R. GRACE & CO.
CLARKSVILLE, MARYLAND
SCALE: 1"=100'
JUNE 1960
MADDOX & HOPKINS • CIVIL ENGINEERS
SILVER SPRING, MARYLAND
147.37 ACRES
REV. 6-15-65

3.6 Special nuclear material may be found in one other location on the site and that location is the storage area in Building 20. This caged area is used primarily for the storage of bird cages of various scrap materials containing special nuclear material and also product in bird cages. The building and the caged area are both locked to prevent unauthorized intrusion.

4.0 DURATION OF LICENSE

The present license SNM-840 expires November 30, 1972. No change is requested in the expiration date by this amendment.

5.0 SPECIAL NUCLEAR MATERIAL REQUIREMENTS

5.1 It is requested that the quantity of uranium 235 in special nuclear material that may be in possession under this license be increased from one kilogram to 200 kilograms.

5.2 The uranium under this amendment may be at any enrichement in the 235 isotope including highly enriched, and it may be found in various chemical compounds throughout the process. These compounds may be solid or in solution as liquids. Neither metallic uranium nor compounds of uranium such as the hexafluoride that are volatile to any significant degree under the conditions that may exist will be in possession.

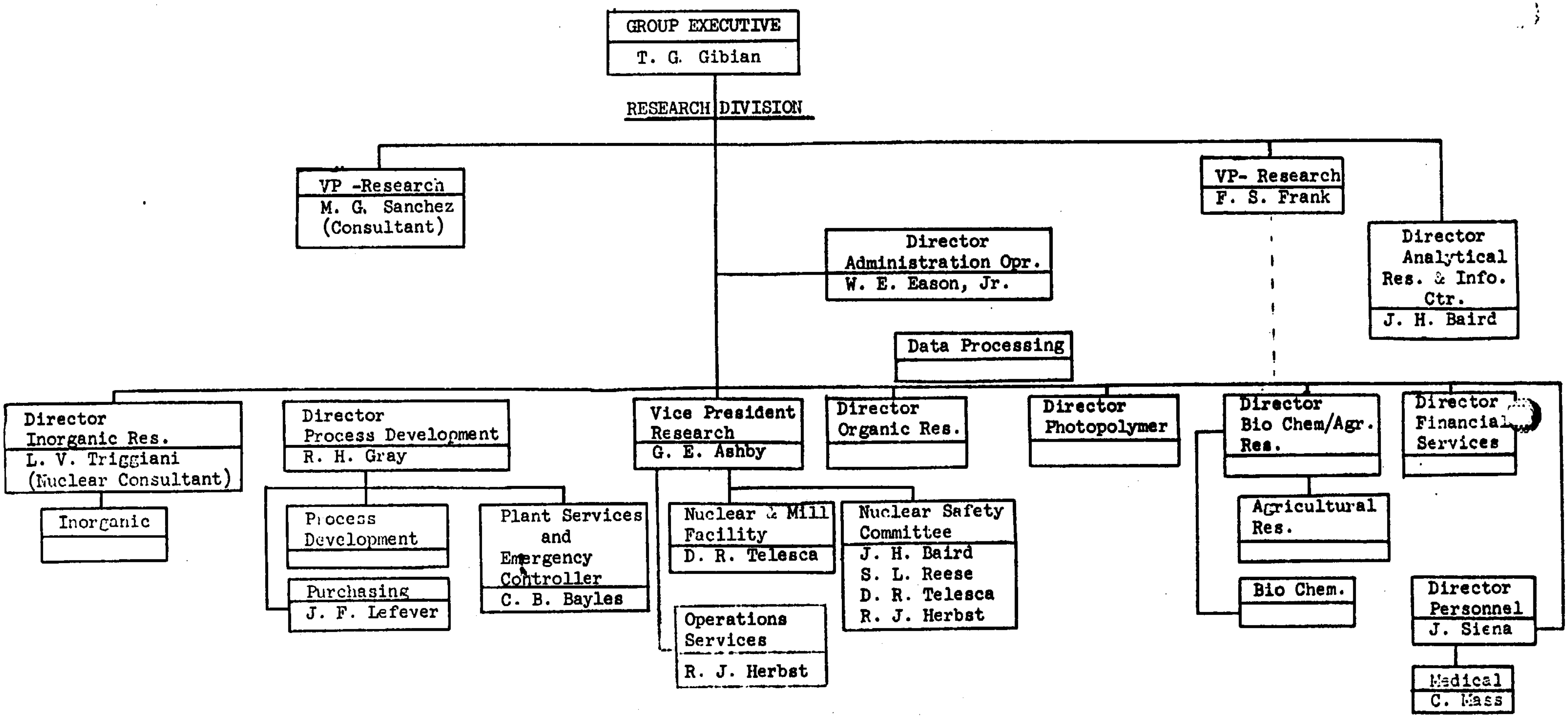
6.0 ORGANIZATION, ADMINISTRATION AND PERSONNEL

6.1 The current organization chart for the Washington Research Center is shown in Figure 6.1. Individuals in the organization who are directly concerned with the nuclear work are listed by name.

FIGURE 6.1

ORGANIZATION OF THE RESEARCH DIVISION

WASHINGTON RESEARCH CENTER--W. R. Grace & Co.



6.2 The Laboratory Director is Dr. T. G. Gibian, a Vice-President of W. R. Grace & Co. who has overall responsibility for all activities at the Center.

6.3 The Manager of Nuclear Operations is Mr. G. E. Ashby, Divisional Vice-President who has overall responsibility for all aspects of the facilities involving special nuclear materials.

6.4 Dr. M. G. Sanchez is a Vice-President of the Technical Group. Dr. Sanchez, who has been very much involved in the nuclear activities at the Center, is a consultant to the Nuclear Operations.

6.5 The Facilities Manager is Mr. D. R Telesca under whom the operations involving special nuclear material are carried out. The organization established to carry out the work is shown in Figure 6.5.

6.6 Nuclear Safety Committee-- In addition to the usual line and staff functions essential to the safe operation in the facility, the Vice President, Manager of Nuclear Operations, has established a Nuclear Safety Committee and he appoints the members to the Committee. This Committee is composed of four members. In the event of illness, or absence, an alternate may be designated.

The members are experienced in the various aspects of nuclear operations involving special nuclear material as follows:

At least one member shall have a degree in the natural sciences or engineering or equivalent* and at least two years of practical experience which shall include responsibility for

A. The nuclear criticality safety aspects of plant operations involving large quantities of fissile materials.

(*Equivalent means 4 years experience = 1 year college)

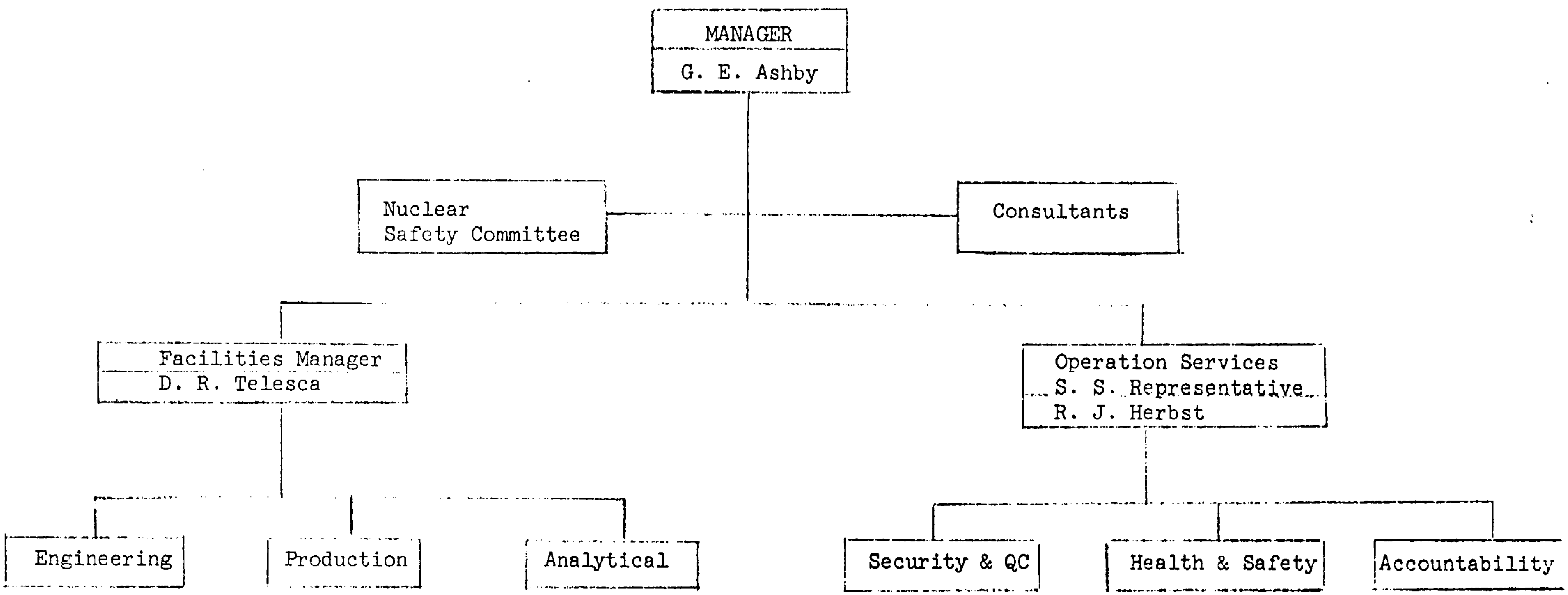
B. The radiation safety aspects of plant operations involving operations with radio active materials.

Nuclear Criticality Safety--At least one member of the Nuclear Safety Committee shall have a degree in the natural sciences or engineering or equivalent where equivalent means 4 years of practical experience in lieu of a year of college. In addition, he shall have at least two years of practical experience which shall include responsibility for the nuclear criticality safety aspects of plant operations involving large quantities of fissile materials.

Radiation Safety--At least one member of the Committee shall have a degree in the natural sciences or engineering or equivalent where equivalent is defined as two years of practical experience in plant operations including radiation safety in lieu of a year of college. In addition he shall have at least two years of practical experience which shall include responsibility for radiation safety aspects of plant operations involving operations with radioactive materials.

Process Safety--At least one member of the Committee shall have a degree in the natural sciences or engineering or equivalent where equivalent is defined as two years of practical experience in chemical processing in lieu of a year of college. In addition he shall have at least two years of practical experience which shall include responsibility for the safe operation of processes of a chemical nature.

ORGANIZATION OF THE NUCLEAR GROUP



As of August 1, 1972, the individuals in these positions are:

V. P. Research - Manager Nuclear Operations	G. E. Ashby
Facilities Manager	D. R. Telesca
Operations Services (SS Representative, Radiation Protection Officer and Accountability)	R. J. Herbst
Process Engineer	J. J. Blouin
General Foreman	C. T. Lamberth, Jr.
Analytical	N. H. Weissert

Consultants

Management & Process	M. G Sanchez
Process	L. V. Triggiani
Administration	W. E. Eason
Process Engineering	H. P. Flack
Criticality & Nuclear Safety	Nuclear Safety Associates S. L. Reese
Accountability & Safeguards	Nuclear Surveillance & Auditing Corporation D. E. George R. Lumb, Alternate

Nuclear Safety Committee

Member	Alternate
J. H. Baird	L. V. Triggiani
S. L. Reese	-
D. R Telesca	J. J. Blouin
R. J. Herbst	D. L. Sillyman

(*Equivalent means 2 years experience = 1 year of college)

C. The safe operations of processes of a chemical nature.

(*Equivalent means 2 years experience = 1 year of college)

D. One member may be chosen for managerial or other skills.

6.6A Operational Committee--There is an operational review board which functions on operational safety, consisting of the Plant Manager and two other members.

6.7 The primary function of the Nuclear Safety Committee is to review proposed equipment change or any proposed operating procedures with regard to process safety, radiation and nuclear criticality safety. The Committee must be satisfied that the procedures and the equipment are safe, and each member or designated alternate must formally approve them by signature prior to use.

6.8 A second function of the Nuclear Safety Committee is to inspect the operations and health and safety records for compliance with approved procedures and pertinent AEC regulations at least once in each month that operations involving special nuclear material are being performed. The Committee will report, in writing, the scope and evaluation of each inspection and its recommendations to the Manager of Nuclear Operations.

6.9 In the event of an accident or emergency involving special nuclear material, the Nuclear Safety Committee will stand ready to aid in the solution or correction of the problem.

6.10 Responsibilities and their Delegation-- The Manager of Nuclear Operations has established the policy that written and authorized procedures for each normal operation involving special nuclear

material shall be prepared and followed. Other emergency situations will receive either verbal or written instructions. This policy is implemented through delegations to appropriate managers and groups responsible for individual aspects of the nuclear activities. Implementation is simplified somewhat by the straight forward operations which are to be performed upon the special nuclear materials and by the limited scope of these operations.

6.11 The Vice President, Manager of Nuclear Operations, who has overall responsibility, has issued directives covering the preparation of Standard Operating Procedures.

6.12 The members of the Nuclear Safety Committee have the responsibility for approving, by signature, changes in procedure or processing to assure adequate protection of personnel and the public from the nuclear hazard and as being in compliance with pertinent AEC regulation and with provision of the amendment. A negative finding by the Committee as to the adequacy of criticality or radiation safety will not be overruled by anyone.

6.13 The Facilities Manager has the direct responsibility for carrying out every operation involving special nuclear material in accordance with the provisions of this license amendment as implemented by approved operating procedures.

6.14 The Vice President, Research has overall responsibility for the Health and Safety program; the security of the facilities as required to safeguard the special nuclear material; and the medical program.

6.15 The Radiation Protection Officer is responsible for carrying out the Health and Safety program. This program includes routine air and smear sampling for contamination and frequent inspection of the operations for compliance with the procedures and the limi-

tations and arrangement of special nuclear materials set forth herein. Action will be taken to correct any deficiency discovered.

Any hazardous condition or non-compliance noted shall be corrected promptly by the operating personnel.

6.16 Technical Qualifications--The Management and staff at the Center are experienced and highly qualified scientific personnel. Members of the staff at the center have had considerable experience in processes involving special nuclear materials over the past decade in the Erwin, Tennessee fuel materials plant, formerly operated by W. R. Grace and its former subsidiary, Nuclear Fuel Services. The knowledge, skill and experience of personnel currently assigned to positions with safety responsibility is presented briefly in the following paragraphs.

6.17 G. E. Ashby - Divisional Vice President, Manager Nuclear Operations,

Education: B.S. Physics

From 1950 to 1964, Mr. Ashby was involved in a number of research projects both before and after coming with Grace. His work in the nuclear field began in 1964 when he became Manager, Nuclear and Ceramic Research. In this position he has been involved in research on oxides, carbides and nitrides of uranium, thorium, zirconium, and the rare earths; microsphere research included not only laboratory operations but in some cases pilot plant and full plant startup activities as well. He also managed programs on other nuclear fuels, nuclear fuel reprocessing, colloidal ceramics, and slurry fertilizer. In 1967 he became Department Director with responsibility for programs of the Inorganic Materials Research Department at WRC. These include major research activities related to: fine size materials, nitrogen

chemistry, catalyst research, and nuclear fuels. Work also includes liaison with the operating division of Grace.

6.18 D. R. Telesca - Plant Manager

Education: B.S. Chemical Engineering, MIT

B.S. Business Administration, Rutgers University

Mr. Telesca joined Davison Chemical Company in July, 1954 as a Chemical Engineer in R&D. He worked as Assistant Manager, Specialty Plant; Plant Superintendent, Specialty Plant, Plant Superintendent Reforming Catalyst Plant. In these positions he had plant level responsibilities for production, including scheduling, raw material delivery and quality, labor scheduling and training.

In 1959 Mr. Telesca transferred to International Metalloids, Inc. (51% owned by W. R. Grace & Co.) in Toa Alta, Puerto Rico with the assignment of Production Manager. He was promoted to Vice President and General Manager responsible for overall operation of producing unit for polycrystalline silicon.

In 1960 he returned to Baltimore and worked at Grace Electronic Chemicals, Inc. (51% owned by W. R. Grace & Co.) as Vice President and General Manager with general management responsibilities for overall operation of silicon monocrystal production.

In 1961 he became Production Manager of the Chemicals Division with management responsibilities for production, maintenance and engineering, process development and quality control for seven producing units of the company.

In 1964 he was assigned to the Division General Management Group Staff with technical and administrative responsibilities on expansion projects.

In 1965 Mr. Telesca was assigned to the Nuclear Development Group, with technical and administrative responsibilities in the area of exapnsion in

the nuclear field. During the major portion of the subsequent period, he was assigned to the Nuclear Fuel Services reprocessing plant at West Valley, New York, to assist in plant completion, start-up and operation. During the past two years, Mr. Telesca has been involved in the process proposed here both at the Erwin Plant of NFS and at the Research Center.

6.19 J. N. Lomonte - Alternate Health and Safety

Education: M.S. Physical Chemistry

Mr. Lomonte has over fifteen years experience in analytical chemistry including a year's experience in the analytical chemistry of uranium enriched in the 235 isotope. His experience in the field of health and safety includes a two year appointment to the Washington Research Center Safety Committee, the second year of which he served as its Chairman. He has also successfully completed courses in Basic Radiological Health and Occupational Radiation Protection given by the United States Public Health Service at the National Center for Radiological Health. Recently, Mr. Lomonte spent three months at the NFS Erwin Plant, where he gained operating and analytical experience with special nuclear materials.

6.20 Leonard V. Triggiani - Director Research - Alternate Safety Committee

Education: Ph.D., Physical Chemistry

Beginning in 1963, Dr. Triggiani became involved in the Nuclear Research Program. He invented a sintering process for microspheric fuels. He did the initial process work on carbide microspheres. During this period, he did considerable trouble-shooting at the Erwin, Tennessee Nuclear Fuels Plant. He also spent four months at the West Valley Plant of NFS working on Np calculations and other reprocessing problems. He invented and developed a process for the manufacture of

UN microspheres. He developed a process for impregnation of PuO_2, ThO_2 and nuclear poisons into urania microspheres and other porous substrates. He developed process conditions for ThO_2 microspheres. He currently directs projects in nuclear fuels, including carbides, nitrides and in plutonium fuels.

6.21 H. P. Flack - Process Consultant

Education: B.S. Chemical Engineering

Since 1962, Mr. Flack has worked almost exclusively on processes dealing with nuclear fuels and primarily with sols and microspheres of various compositions. During this period he was assigned for extended periods to the Erwin, Tennessee Nuclear Fuels Plant of NFS in the development and start-up of processes involving special nuclear material.

6.22 J.J. Blouin - Process Engineer

Education: B.S. Chemical Engineering

From June to September 1964, Mr. Blouin worked as a chemical engineer in the Nuclear Power Division at the Portsmouth Naval Shipyard where he was involved in coolant control in the primary loop of nuclear reactors.

From September 1964 to September 1968 he was on active duty with the U.S. Air Force as a Radiochemist with the 1155th Technical Operations Squadron. During this time research was performed in the areas of radiochemistry, atomic absorption and thermogravimetry. In February 1968 he was appointed to the position of section head in charge of two professional chemists and twelve technicians. In March 1968 was promoted to the rank of Captain.

Mr. Blouin joined the Research Division of Grace as an Assistant Research Engineer in October 1968, where he has most recently been involved in the rate of production of selected particles of UO_2 and metal carbides.

6.23 C. T. Lamberth - Foreman

Education: High School Graduate

Charles T. Lamberth joined Davison Chemical Co, which is a division of W. R Grace & Co., in October, 1951 as a Laboratory Technician in R&D. He worked as a technician on catalyst research, electrodialysis of various sols, etc. until 1960. He transferred to Grace Electronics in 1960 as a Process Supervisor until 1961. Transferred to R&D in 1961 as a Technician Specialist, primary work concerned with auto exhaust catalyst until 1963. Mr. Lamberth worked in the Nuclear Research Laboratory at WRC from 1963 until present. Work concerned mainly with sol and microsphere technology in development room of Nuclear Dept. From 1963 until 1968, intermittent assignments were carried out in the Nuclear Fuel Facilities at Erwin, Tenn. working with production quantities of depleted and enriched uranium. Served as Operator, Shift Foreman and Quality Control Technician at NFS in those years. Mr. Lamberth was promoted to Research Associate and Security Officer at W. R. Grace & Co. in 1968. He is presently assigned to the Nuclear Fuel Facility, Bldg. 16-A, Washington Research Center, as Foreman in charge of development quantities of nuclear fuels.

6.24 D. L. Sillyman - Health & Safety Technician

Mr. Sillyman worked in the Health Physics Division Radiation survey at Oak Ridge National Laboratory for a year prior to coming to W. R. Grace in 1968.

Before that he was employed in the Health Physics section of the Martin Marietta Corporation Radioisotopes Production facility at Clearfield, Pennsylvania for 4.5 years. Megacurie quantities of Sr^{90} were encapsulated under contract to the AEC at that facility.

Prior to his assignment at Clearfield, he worked in the Health Physics section of Martin Marietta at the Baltimore plant for six years. Experience at the Baltimore plant included monitoring during fabrication of enriched uranium reactor fuel assemblies.

6.25 H. P. Patterson - Analytical Chemist

Education: B.S. Chemistry

From January 1960 to May 1963, Mr. Patterson was employed at Nuclear Science and Engineering Corporation, Pittsburgh, Pa., as an analytical chemist. Duties included analyses for nuclear fallout in rainwater, food, human tissue, etc. Also performed analyses of nuclear reactor water and reactor corrosion products for the quantitative detection of various radioactive constituents. Performed on site radiochemical analyses associated with the reactor startup of the N.S. Savannah and a nuclear sub.

He was employed at Westinghouse Bettis Atomic Power D vision from June 1963 to December 1968 as a radiochemist. From June 1963 to December 1965 he was the Bettis representative chemist at the test loops at the National Reactor Testing Station in Idaho Falls, Idaho.

Duties involved control chemistry of all test loops, startup chemistry, loop decontamination and analysis of test loop coolant for radionuclides to verify integrity of test specimens.

From January 1965 to December 1968 was located at the Bettis Laboratory in Pittsburgh. Duties involved analysis of reactor fuel rods for burn-up of fissionable material. Developed procedures for lab analysis and for nuclear sub use. Performed separations of rare earths materials. Wrote sections of standard lab manual. Was sent out periodically to shipyards to perform radiochemical analysis involved in startups of new or refueled nuclear subs and special testing.

He is currently developing methods of analysis for trace elements in urania microspheres. Wrote up the procedures used for cerium analysis by amperometric titration.

6.26 R. J. Herbst - Operation Services

Education: B.S. Ceramic Engineering, University of Baltimore
Ph.D., Ceramic Engineering, University of Illinois

Dr. Herbst's nuclear experience begins in 1955. During the summer of 1955, '56 and '57, he was employed in the Ceramic Group, Metallurgy Division, Argonne National Laboratory as a Jr. Research Technician. Upon completing graduate work in fulfillment of the requirements for his PhD in 1963, Dr. Herbst joined the Research Division of the Westinghouse Electric Corporation. In 1964 he transferred to what is now Westinghouse Nuclear Fuel Division.

Dr. Herbst conducted research on the fabrication of low-enriched uranium fuels for power reactor applications in the Nuclear Fuel Divisions, Materials Systems Laboratory. His experience covers most

aspects of fuel manufacture from UF_6 conversion to fuel element assembly. He participated in the design of the MSL's, Plutonium Development Laboratory. In 1967, he was named Special Nuclear Materials License Administrator responsible for assurance of compliance with all aspects of MSL's SNM license including materials accountability and nuclear and radiation health safety.

In 1968, Dr. Herbst joined the Nuclear Department of W. R. Grace & Co's Research Division. At Grace he has continued fuel research gaining added experience in chemical operations with highly enriched uranium.

6.27 N. H. Weissert - Analytical Chemist

Education: B.S. (Chemistry) Grove City College, Pennsylvania

From 1951 to 1960 worked in all phases of Analytical Research at Gulf Research and Development Co. Last 3 years were spent doing method development in gas chromatography.

From July 1960 to 1963 joined Quaker State Oil Refining Co. to modernize research laboratory. Acquired analytical and other research instrumentation; trained laboratory and plant personnel in their use.

On December 1963, he joined WRC as a research chemist in the Analytical Research Department. Developed analytical procedures and analyzed samples, using gas chromatography. Mr. Weissert served as analytical liaison on NTAN project at WRC and Hampshire Chemical Co.

In 1967, he moved into x-ray spectrochemical group to learn fundamentals of technique. Began work on method development using x-ray spectrochemical techniques.

In November 1967 he went to Davison, Curtis Bay to set up a new x-ray vacuum spectrometer. Began analyzing samples in January 1968 after

calibration of equipment. Responsible for x-ray group at Curtis Bay including training and acquiring additional personnel. Interfaced procedures developed at WRC into Curtis Bay plant.

In July 1968, he returned to WRC from Curtis Bay assignment.

In December 1968, Mr. Weissert was promoted to Research Supervisor of Established Methods Laboratory, X-Ray Spectroscopy and Gas Chromatography Groups.

Mr. Weissert has used tracer techniques on several projects and has had experience in neutron activation. He used x-ray diffraction and spectroscopy in support of nuclear projects.

He was assigned to the Nuclear Facility as Analytical Chemist in October, 1971.

6.28 C. S. Mass - Medical Director

Education: M.D. Howard County Medical Center

Post graduate courses completed by Dr. Mass include the following: Resuscitation, New York 1952 (Dr. Flagg); Physiological Basis of Medicine 1955, University of Maryland; Pulmonary Physiology and Pathology 1963, Hahnemann Medical College, Philadelphia.

He is Attending Physician in Medicine, Bon Secours Hospital, Maryland; Member of the Advisory Board Committee for Nurse Education Program at Howard Community College in Maryland.

At. W. R. Grace & Co. he is responsible for pre-employment and executive personnel examinations since 1959. Dr. Mass became the W. R. Grace & Co. Medical Director in 1972.

6.29 J B Ziegler - Medical Advisor

Education: M.D. University of Maryland Medical School

Post graduate courses completed by Dr. Ziegler include the following:

Nine month course--Neuro-Physiology, Tulane Medical School, 1952.

Nine month course--Neuro Pathology, Tulane Medical School, 1953.

Basic Course--Clinical Hypnosis in Medicine, American Medical Hypnosis Society, Philadelphia, Pa., January 1961.

Advanced Course--Clinical Medicine, University of Md. Medical School (one credit).

American Academy of General Practice, Hagerstown, Maryland, November, 1962.

Advanced Course--Clinical Hypnosis in Medicine, American Medical Society of Hypnosis, Philadelphia, Pennsylvania, January 1963.

Training Seminar-Medical care and treatment of radiation accidents Brookhaven National Laboratory, October 23, 1968.

Dr. Ziegler has been Medical Examiner and Counselor, Member of

National Council, Boy Scouts of America since 1954. He was principal physician at Brooke Grove Chronic Disease Hospital, Olney, Maryland and principal physician at Sharon Nursing Home, Sandy Springs, Md. during the period 1954-58. He was Chief of Neurology Service (Out-Patient Clinic and Teaching House Staff), Suburban Hospital, Bethesda, Maryland from 1958-63.

From 1958 to 1962 he taught Clinical Neurology--Residency and Internship at Suburban Hospital; Physical fitness and its need in the military to Navy and Air Force Pentagon Staff, Physical medicine (new concept) to the Presbyterian Hospital Staff, New York City; and he also held the position of Assistant Professor--Physical Medicine, Georgetown University School of Medicine.

6.30 E. A. Kerns - Registered Nurse

Mrs. Kerns attended numberous in-service education programs relating to radiation hazards, First Aid Techniques, emergency procedures and evacuation techniques while on duty with the Nurse Corps, United States Navy.

She is a qualified instructor in Cardio-Pulmonary resusitation with the Heart Association of Maryland

During the past three years at W. R. Grace, she has become familiar with the schedule of analysis for employees involved in working with radioactive materials.

At present, Mrs. Kerns is reviewing the course manual for "Radiological Health for Nurses" published by the Division of Radiological Health, U.S Department of Health, Education and Welfare.

This manual was designed to provide nursing personnel with the fundamentals of radiation, its biological effects, its medical applications, its nursing implications and the basic principles of protection.

70-456

For Div. of Compliance

SECTION 7
COPY No. 5

APR 9 1970

~~GRACE PROPRIETARY INFORMATION~~

USAEC

Not ~~to be released outside~~ nor ~~to be duplicated, used or disclosed to others~~ without the express permission in writing of W. R. Grace & Co.

1000

7.0 PROCESS EQUIPMENT

This document contains paragraphs 7.1 through 7.23 (pages 22 through 31, and Figures 7.3 a and 7.3 b of the application.) Also included herewith is Drawing E-69020-51, which is a part of Appendix B.

TABLE OF CONTENTS

6.31 Stanton L. Reese - Consultant

Mr. Reese has been in the nuclear field for eighteen years. For the past four years he has been a Partner in Nuclear Safety Associates. During the previous four years he took part in the NFS project which led to the commissioning of the West Valley reprocessing plant. During this period he served as Assistant to the President of NFS and to the West Valley General Manager. He was responsible for and performed the entire criticality analysis for the initial plant design and he prepared the criticality section of the Safety Analyses.

In 1956 Mr. Reese was one of three principals who put the W. R. Grace & Company into the nuclear field by designing, constructing and placing in operation a commercial feed materials plant in Erwin, Tennessee. During the six years Mr. Reese was at Erwin, he held positions of Technical Director, Production Manager, and Plant Manager. He did the criticality analyses and the licensing for the plant processes and shipping containers. He was instrumental in the development, plant design and construction, and the production of a variety of products and services. Mr. Reese entered the nuclear field in 1951 at the Atomic Energy Commission's Feed Materials Production Center at Fernald, Ohio. At FMPC he was the Department Supervisor of the Metals Pilot Plant. While there, he became involved in nuclear criticality safety and established safe procedures for operations with enriched uranium under the tutelage of Dr. D. Calihan of ORNL.

Mr. Reese is 45. He did his undergraduate work in Chemical Engineering at the University of Cincinnati and at the University of Alabama the latter as a part of his service in the U. S. Army during World War II. During that service he earned three battle stars in Germany and was decorated with the Silver and Bronze Stars. He left the service with the rank of Captain. He is a member of ANS, AIF, INMM, ASM, and AIMME. He has been and is active on numerous working committees of each.

6.32 Training--Prior to working with fissile materials, each employee who is to work in the nuclear area will participate in a basic course covering nuclear criticality safety, radiation safety, and the use and maintenance of safety equipment. This will be followed by an examination.

These will be followed by periodic safety meetings on pertinent safety matters including the above.

The part of the basic instruction in safety dealing with nuclear criticality will define the chain reaction, how it is sustained, and describe the mechanisms used to prevent a criticality accident. This instruction will be prepared and presented by the criticality member of the Nuclear Safety Committee, or his alternate.

The radiation safety instruction will include a definition of the types of radiation of concern; exposure mechanisms; operating practices to minimize exposure, and the methods used to determine exposure such as film badges, air sampling, area monitoring and biological analyses. Records will be discussed. This portion of the instruction will be prepared and presented by the radiation safety member of the Nuclear Safety Committee, or his alternate.

Mr. Reese is 45. He did his undergraduate work in Chemical Engineering at the University of Cincinnati and at the University of Alabama the latter as a part of his service in the U.S. Army during World War II. During that service he earned three battle stars in Germany and was decorated with the Silver and Bronze Stars. He left the service with the rank of Captain. He is a member of ANS, AIF, INMM, ASM, and AIMME. He has been and is active on numerous working committees of each.

6.29 Training--Prior to working with fissile materials, each employee who is to work in the nuclear area will attend a four hour basic course covering nuclear criticality safety, radiation safety, and the use and maintenance of safety equipment. These will be followed monthly with safety meetings, each of about one hour duration, on pertinent safety matters including the above.

The part of the basic instruction in safety dealing with nuclear criticality will define the chain reaction, how it is sustained, and describe the mechanisms used to prevent a criticality accident. This instruction will be prepared and presented by the criticality member of the Nuclear Safety Committee, or his alternate.

The radiation safety instruction will include a definition of the types of radiation of concern; exposure mechanisms; operating practices to minimize exposure, and the methods used to determine exposure such as film badges, air sampling, area monitoring and biological analyses. Records will be discussed. This portion of the instruction will be prepared and presented by the radiation safety member of the Nuclear Safety Committee, or his alternate.

Following this basic instruction period, each employee will be instructed in the part of the operation he is to perform. He will be given sufficient

time to completely familiarize himself with the written procedure for that operation. He will then carry out the operation with the help of his foreman who will explain how nuclear criticality and radiation safety are provided for in the operation. His understanding of both the operation he is to perform and the safety features involved will be determined by oral examination by the General Foreman. When the General Foreman is satisfied that the employee has demonstrated the capability of performing the operation satisfactorily and safely he will so indicate in the job qualification record. The employee may then perform the operation.

In view of his importance in the training program the General Foreman shall have received a degree in the natural sciences or engineering or its equivalent, where equivalent is defined as two years of practical experience in plant operations in lieu of a year of college. In addition, he shall have at least one year of practical experience in the nuclear field.

7.0 PROCESS EQUIPMENT

Paragraphs 7.1 through 7.23 and Figures 7.3 a and 7.3 b (pages 22 through 31) contain proprietary information and are to be formed in a separate document.

7.0 PROCESS EQUIPMENT

7.1 The process description presented in this section contains proprietary information, and it is desired that the first 21 paragraphs of this Section 7 be withheld from public inspection as permitted by 10 CFR 2.790 (b).

7.2 This section of the report describes the process to be used, the salient safety features of the equipment in which the process will be carried out, and the various auxiliary equipment, instruments, and supplies which are important to safety.

7.3 Description of the Process-- The proposed process consists of the following major steps:

1. Dissolution of UO_2 or U_3O_8 in HCl,
2. Preparation of a UO_2 sol by dialysis,
3. Formation of UO_2 particles by dripping the sol in discrete droplets through a dehydrating solvent,
4. Drying and sintering the particles,
5. Coating the particles.

These steps are shown in Figure 7.3a and Figure 7.3b and are described in more detail in subsequent paragraphs.

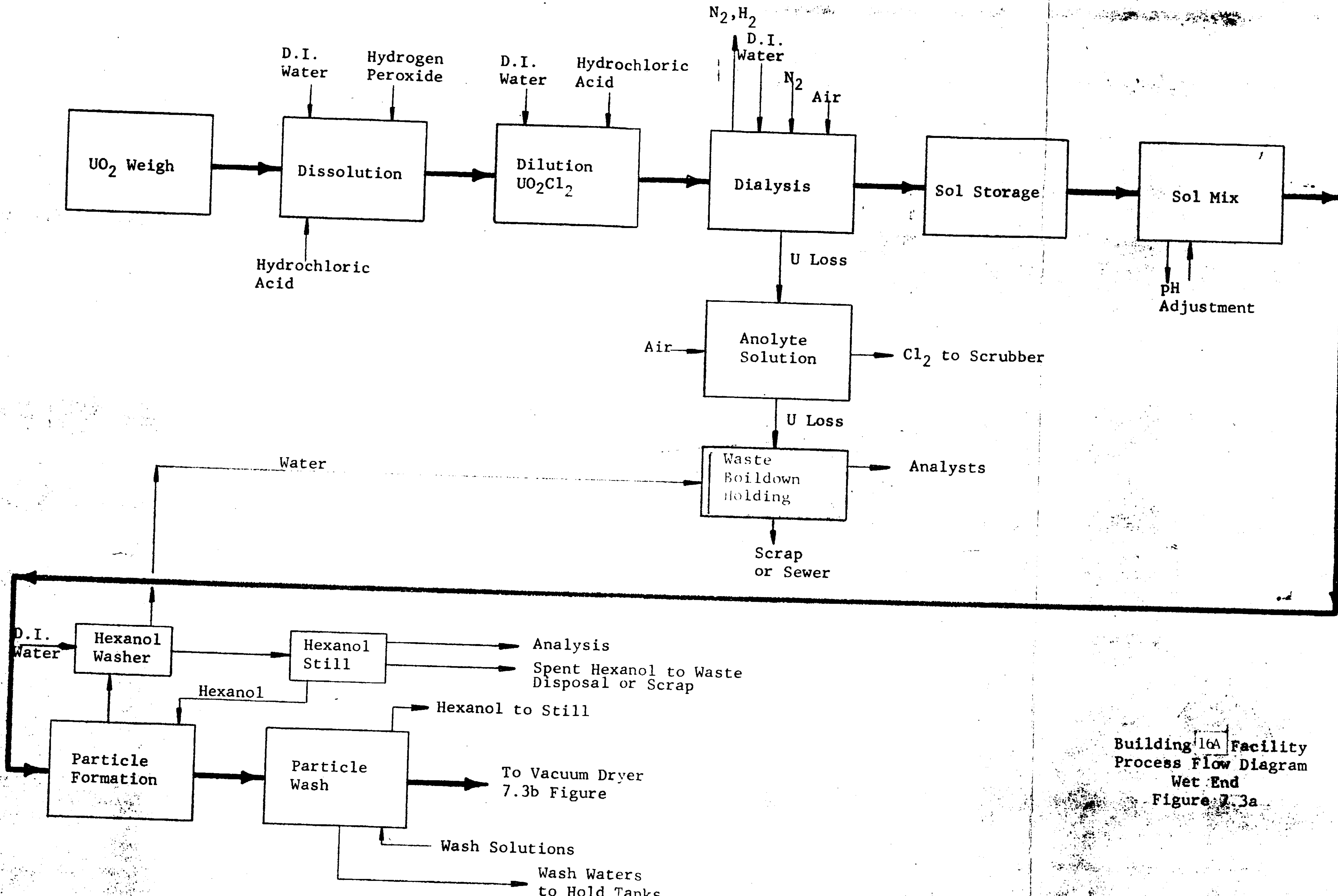
7.4 Receipt, Sampling and Storage-- The raw material for the process will be purchased from vendors in the form of enriched uranium oxide (UO_2 or U_3O_8). This oxide will be received in DOT approved packages. The amount of uranium oxide in any single container within the package will preferably be an amount convenient to use as a single

FIGURE 7.3a

BUILDING 16A FACILITY

PROCESS FLOW DIAGRAM

WET END



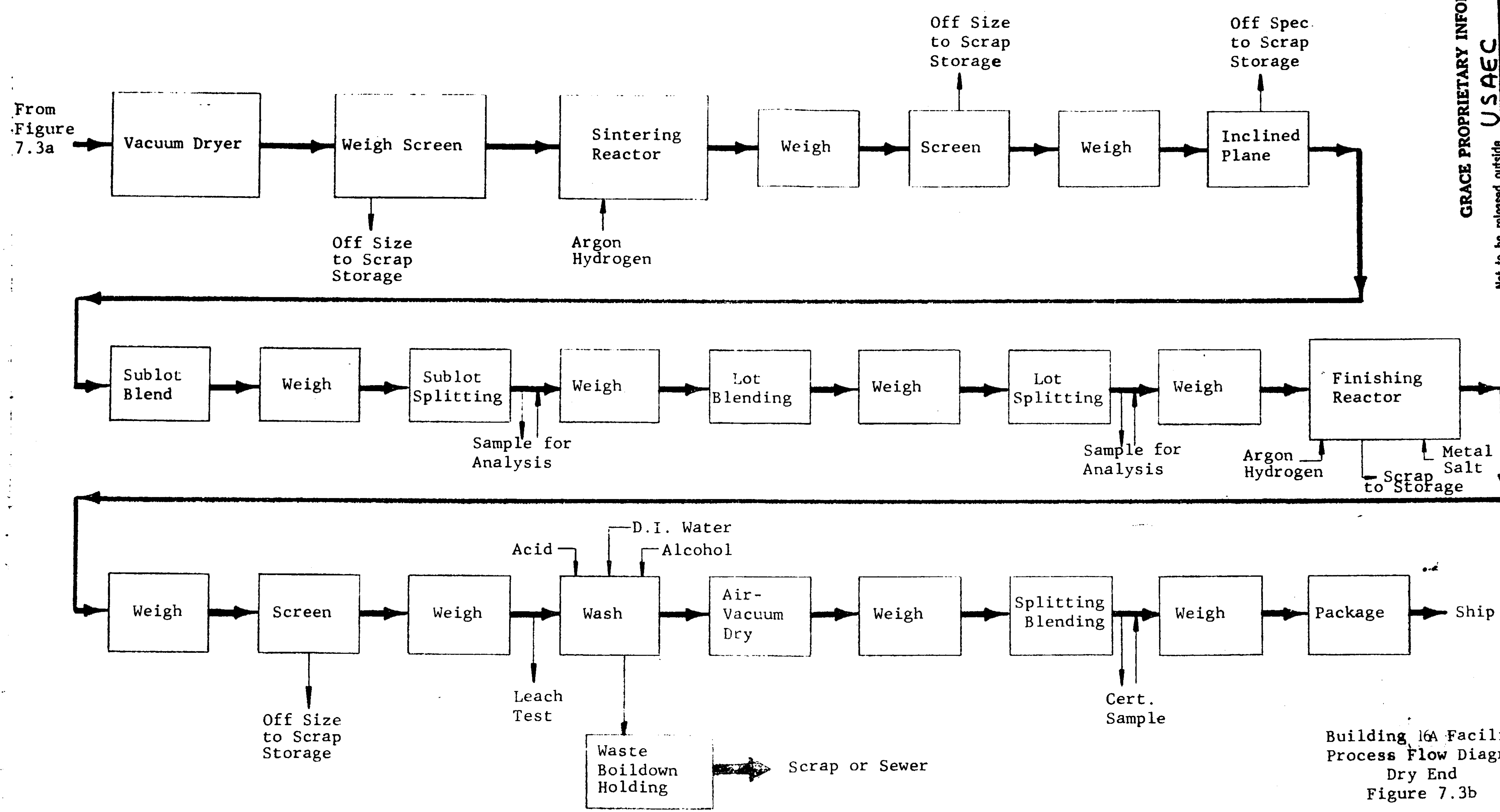
Building 16A Facility
Process Flow Diagram
Wet End
Figure 7.3a

FIGURE 7.3b

BUILDING 16A FACILITY

PROCESS FLOW DIAGRAM

DRY END



Building 16A Facility
Process Flow Diagram
Dry End
Figure 7.3b

dissolver batch. Upon receipt, the bird cages will be taken to the authorized area and locked. As soon as practical after receipt, the containers of uranium oxide will be taken, one at a time, from the birdcages and the gross weight of each container will be verified. Samples will be taken to the laboratory for analysis. When processing is to begin, the quantity required for a dissolver batch will be weighed out in dry box 14-1 and emptied into the dissolver. As each receiving container is emptied, its tare weight will be verified. Thus, a check will be obtained upon the vendor's shipping weights.

7.5 Dissolution--Dissolution of the uranium oxide is carried out in a vertical, cylindrical dissolver 13-1 of four inch Schedule 40 steel pipe 30 inches long. This pipe has a 5/64 inch thick teflon liner on all internal surfaces and it is surrounded by a six inch Schedule 40 water jacket. The outer surface of the water jacket is bare and not subject to close reflectors. As an alternate a bare walled five inch Schedule 40 pipe 40 inches long may be used to replace the above vessel. A separate 2 inch diameter glass heat exchanger is used. It is located above filter 06-2B which is 30 inches from the dissolver.

7.6 The pre-weighed uranium oxide charge is added to the dissolver through a removable head. Dissolvent and oxidizing agent are metered into the tank. The mixture is stirred by means of an impeller during the course of dissolution, or in the alternate arrangement it is circulated through the glass heat exchanger. Temperature is controlled by regulating the amount of steam admitted to the heating coils in the secondary heat exchanger 02-1 in this closed heating and cooling system. A positive pressure of 20 psig is maintained on the heating-cooling system thus a leak would push liquid into the dissolver. This

pressure is recorded on the run sheet before each use, and the level of liquid in the expansion tank will also be recorded. If either is below the established level the unit is examined for a possible leak before use. A small quantity of gases and vapors are emitted and they are filtered before being sent to the exhaust system. The final uranium solution is cooled and drawn off through a filter (06-2A or 2B) into a dilution tank 10-3A, 3B, or 3C which is made of five inch PVC pipe jacketed with a steel sleeve to prevent expansion in the event that hot solution is inadvertently drained from the dissolver.

pressure is recorded on the run sheet before each use, and the level of liquid in the expansion tank will also be recorded. If either is below the established level the unit is examined for a possible leak before use. A small quantity of gases and vapors are emitted and they are filtered before being sent to the exhaust system. The final uranium solution is cooled and drawn off through a filter (06-2A or 2B) into a dilution tank 10-3A, 3B, or 3C which is made of five inch PVC pipe jacketed with a steel sleeve to prevent expansion in the event that hot solution is inadvertently drained from the dissolver.

The solution in which the uranium is in the form of UO_2Cl_2 is adjusted in uranium and acid concentrations in the dilution tank to that desired for the next step (dialysis).

7.8 Dialysis--The adjusted UO_2Cl_2 solution is next transferred on a batch basis to the dialysis feed tank 10-4 which is a five inch diameter PVC tank, to which is attached a sight glass to permit observation of the color of the solution. The solution is then circulated to one side of the dialysis cell 13-2. The dialysis cell is separated into two compartments by a semi-permeable membrane. Water is circulated through the second (anolyte) side of the cell. In this cell, the chloride ion is essentially removed from the UO_2Cl_2 leaving the UO_2 in the form of a sol. All of the chloride goes off as chlorine gas which is scrubbed out in NaOH in tank 10-8. The spent caustic solution normally contains no fissile material and it is sent to holdup tank 10-42A or 42B. Some uranium does find its way into the anolyte as noted in Figure 7.12. The resulting solution is pumped from anolyte tank 10-7 to boildown feed tanks 10-37A or 37B. Hydrogen gas is also given off from the dialysis cell. This gas is diluted with sufficient air in the aerator 10-5 to lower the hydrogen concentration well below the lower explosive limit. This mixture is vented to the plant ventilation system--where additional dilution of the hydrogen will occur placing it even farther below the explosive limit. The dialyzed sol is collected in storage tanks 10-10A or 10B made of five inch PVC pipe.

7.9 Particle Formation--The sol is pumped from the storage tanks into the column feed tank 10-11. This mixture is then continuously circulated through constant overflow tank 10-12. A stream from this tank is continuously fed to the additive mixer 05-3 followed by the column feed tank 10-13A or 13B and thence into one of the particle formation columns 13-6A or 6B through a series of hollow needles at the top. The droplets of feed are injected into a counter-flowing stream of hexanol, introduced at the bottom of the unit. In falling through the hexanol, the sol is dewatered and solid UO_2 particles are formed. These are collected in the conical bottom section cylinder. Hexanol flows out the top of the unit to solids filter 06-7A or 7B and then to washer 10-18 where it is continuously washed with water. It then passes through the backup hexanol-water separator 10-19 and then to the still 09-1 to remove excess water. It is then recycled to the unit.

7.10 The particle formation units 13-6A and 6B are simply long columns of six inch diameter schedule 40 pipe on approximately 36 inch centers. They have a tapered section at the bottom so that the bottom outlet is two inches in diameter. The particles are periodically drained into a four inch diameter beaker and transferred by hand to the washing units.

7.11 Drying and Sintering--The particles may be removed from the bottom of the formation column in one of two ways. The particles are periodically removed in batches of a few hundred grams by drawing off into a beaker. The beaker is moved to the wash stand. This stand consists of a series of 4 inch diameter glass pipe, wash bottles 12-44A-44H twelve inches long which fit into fixed holders all located in the same plane and thus during the washing operation these units are fixed at 24 inches apart on centers. A batch of green particles is transferred to one of these washer units which has sintered metal filters at both ends to retain the particles.

Alternatively, the beaker of particles may be removed from the formation column to the first pipe tank of a washer unit consisting of a series of short vertical pipes with diameters up to four inches. These pipes are in one plane with 18 inches between centers. The hexanol with the particles is replaced by the addition of aqueous ammonia. The hexanol overflows to the same washing system solution collector as do the movable wash bottles. Ammonia solution is pumped through the initial tank from the bottom which overflows into the bottom of the second tank of different size which in turn overflows into the bottom of a third tank. The large particles remain in the first tank. The specification particles collect in the second tank and fine particles collect in the third tank. The solution reservoir for this system is 4 inches in diameter. The oversize and undersize are returned directly to dissolution. The on size particles are drained into the present three inch diameter furnace tube which is transferred to its furnace. One washer unit is provided for each particle formation column.

7.12 The hexanol that is removed is permitted to separate from the aqueous solution in 6 inch diameter pipe tanks 10-24 through 10-26. The aqueous solutions are routed to disposal by way of the waste hold tanks 10-42A or 42B and the hexanol that has separated out is pumped

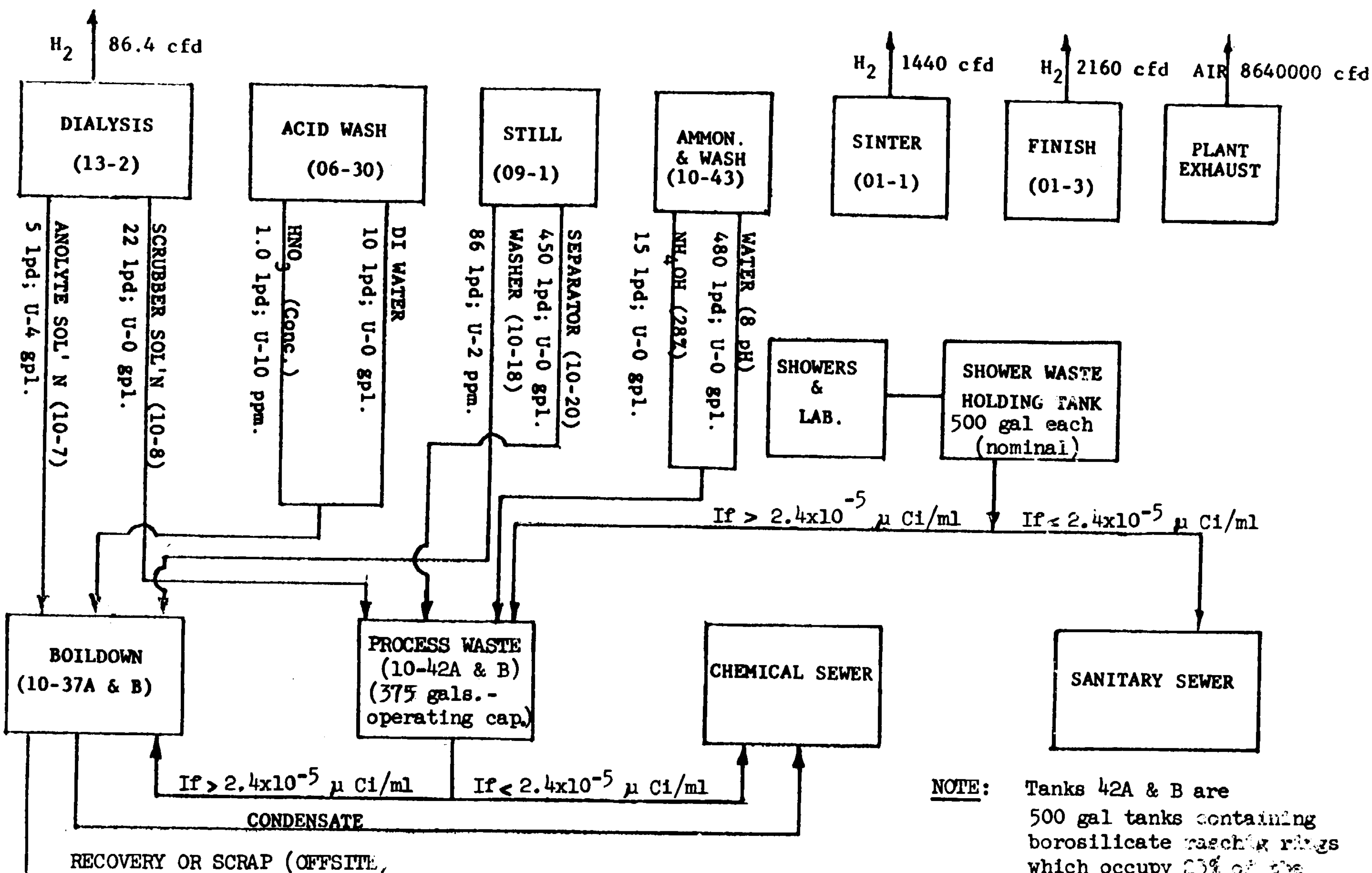
GRACE PROPRIETARY INFORMATION

Not to be ~~released outside~~ USAEC and/or ~~NSA~~ nor to be duplicated, used or disclosed to others without the express permission ~~in writing of W. R. Grace & Co.~~

WASTE STREAM FLOWSHEET

Basis: 24 Hr. Operating Day

Figure 7.12



NOTE: Tanks 42A & B are 500 gal tanks containing borosilicate raschig rings which occupy 20% of the space.

lpd = liters per day; gpl = grams per liter; cfd - cubic feet per day

Docket No. 70-456 Date _____ Revision No. 2 Date 7/29/70 Page 25B

to spent solvent tank 10-22. The wastes from the entire process and their treatment is shown pictorially in Figure 7.12.

7.13 When the glass wash bottle system is used the drained bottle containing the particles is removed from the stand and carried to the table on the opposite side of the vacuum dryer 08-7 where the particles are poured into a metal tray not greater than 8-7/8 inches long by 5 inches wide by 1 inch deep. Each tray is filled to a depth of about 3/4 inch. The cavity in the drying oven is such that a maximum of six trays may be placed in the oven at one time. The oven is evacuated and heated to about 250°C to assure complete removal of moisture from the particles. The dried particle density may approach 35% of theory for UO_2 or 3.8 g/cc. The U-235 density with random packing in the tray could then approach 2.1 g/cc. Average bulk density experienced is 1.5 g/cc. The dryer is cooled and a cover is placed upon each tray of dried particle which is then carried to the gove box transfer station 14-3. When the tube furnace is used for drying, the aqueous solution is removed through the screened end which is valved to vacuum through a filter which removes particulate. The solution is routed as described in paragraph 7.12. The furnace is then heated to about 250°C. After cooling it is removed to transfer station 14-13, as are the trays described above. In this box the material is transferred to a standard canister.

7.14 The standard canister is then moved to the weigh station where it is weighed. These canisters measure 3.5 inches outside diameter by 9 inches in length. This weigh station is also used for weighing the sintering charge, the sintered product and the physically defective material separated in the adjoining glove boxes. These boxes are all maintained under inert atmosphere. Water is not permitted nor is there any mechanism by which it can enter this system of boxes.

7.15 The standard canister with a three kg sintering charge in it and a furnace charging cover on it is moved to the sintering furnace 01-1 which is a vertical pipe not greater than 3 inches in diameter. The contents of the container are emptied into the furnace and the particles are subjected to high temperature in the presence of hydrogen. The furnace is allowed to cool under inert gas and the densified particles are removed from the furnace into a storage canister using reactor unloading device 04-11. They are then returned to the weigh station.

7.16 Screening, Sampling, and Blending--At this point approximately 250-gram quantities of the particles are weighed into stainless steel beakers. Material is removed from the glove boxes in standard canisters only.

7.17 One at a time, charges of particles are placed in a set of eight inch diameter Rotap screens and the product is sized. Off-size material is bottled and transferred to scrap or recycle storage. Material which meets product size requirements is next tested for shape on an inclined plane. Again product which does not meet specifications is bottled and sent to scrap or recycle storage. Samples are taken of material which has met all of the size and shape requirements to determine the density of the product.

7.18 A number of individual batches of satisfactory product are next blended and split, and additional samples taken. Larger lots are again blended, split, and sampled, and finally bottled in the standard canister in approximately three kg lots, for storage until finishing is to be carried out. All particle blending is performed in a blender which is bare wall and housed in a glove box under inert gas. All of the glove boxes have cartridge type replaceable filters on the exhaust outlet.

7.19 Finishing the Particles--Next approximately three kg batches of UO_2 particles which meet product specifications are placed in the finishing reactor 01-3. This unit, which is a tube not greater than three inches in diameter is about forty inches long and is installed inside ~~a furnace~~. The particles are fluidized with a mixture of argon and hydrogen. Part of the hydrogen passes over

heated metal chloride, vaporizing it and carrying the metal chloride into the fluid bed reactor where the chloride is reduced, depositing a thin metal coating on the UO_2 particles. The exit gases next pass through a glass disengaging section to remove any particles which may be entrained. The gases then pass through a filter to remove fine solids that may be entrained. The duplicate filters 06-22A or 22B contain paper cartridges 2-3/4 inches in diameter which collect particles 0.08 microns in diameter or larger. They are enclosed in a 3-7/8 inch by 1/16 wall stainless steel container 10-1/2 inches long. The filters are used alternately. The quantity of fissile material collected per unit is normally extremely small. The gases then pass through a back siphon trap for the neutralizing solution and then they are bubbled through the sodium hydroxide neutralizing solution to remove HCl from this off gas stream.

7.20 Screening and Washing--The coated particles are then cooled and transferred in one of the standard canisters to a screening station 14-8 where again off-size particles are separated out, bottled, and removed to scrap recovery storage Particles which fall within the acceptable range are transferred to a washing station in glove box 14-9. Herein the particles are washed first with acid, then with water, and finally with ethyl alcohol. The washing solutions pass through filter 03-31 on their way to 10-36, the waste holdup tank. The washed particles are placed in a tray not greater than one inch deep and placed in an oven where they are dried under vacuum.

7.21 Blending, Sampling, and Shipment---The dried coated particles are put through a splitter in glove box 14-11 and then sampled for final product acceptability tests. The finished product will be stored in safe storage rack 11-14 and it will finally be placed in packages approved for the shipment of this material. Various lots are blended, split again, sampled for final retention samples, and then weighed out into final shipping bottles. The amount of U-235 per shipping bottle will be recorded and checked by the supervisor and will not exceed 11 kg. The loaded shipping bottles are stored in their birdcage shipping packages.

TABLE 7.2

RADIATION PROTECTION EQUIPMENT

FUNCTION	EQUIPMENT PROVIDED	NUMBER	LOCATION	REMARKS
To Sample Air:				
Continuous Air Sampling	Air Sampler, MCI Inc., Model HD-28 Constant Flow Sampler with Gast Pump and Gelman Filters	3	1-in Lab Metallographic Area 1-in Process area near 14-2 1-in Mezzanine near 14-1	Sample Head at breathing level in each room
Spot Air Sampling	Air Sampler, Staplex Model TFIA, Hi-volume	1	Portable	
Final Effluent Air Sampling	Gast Pump and Gelman Filter	1	In exit duct down-stream of final filter	Samples taken intermittantly
Counting Air Filter Samples	Liquid scintillation counter Nuclear-Chicago Mark 1--Model 6860	1	Laboratory in Building 2	Samples counted daily
To Survey Surfaces:				
Monitoring Floors	Alpha Floor Monitor, Eberline Instrument Corp. Model FM-3G, 3 Scale 10^3, 10^4 and 10^5 cpm	1	Portable	
Monitoring Other Surfaces	Alpha Survey Meter, Eberline Instrument Corp. Model PAC-3G 0-1000 cpm Alpha	1	Portable	
	Survey Meter, Victoreen Instrument Co. Model Thyac III, with Alpha, Beta, Gamma Probe 0-8 cpm Alpha, 0.2,2,20 and 200 mr/hr scales for Beta-Gamma	2	Portable	Used for more inaccessible areas and if Beta-Gamma Survey should be required

Docket No.______________ Date______________ Revision No.______________ Date______________

TABLE 7.23 (con't.)

RADIATION PROTECTION EQUIPMENT

FUNCTION	EQUIPMENT PROVIDED	NUMBER	LOCATION	REMARKS
To Monitor Liquid Effluents:	Holdup Tank	2	Mezzanine level and ground floor	Samples counted on liquid scintillation counter estimated level of detection 3 CPM
To Detect Criticality Event: Alarm	Victoreen Instrument Co. Model 715 detectors or other approved sensors	3	1-in wall between lab rooms 1-in North wall with view of entire process 1-in Bldg. 20 storage	Alarm loud enough to alert entire Research Center
	Model 712 Control Panel with Evacuation Alarm or similar approved system	1	Control Panel at entrance in room 10; Claxon outside building	
Record Event	Eberline Instrument Co. Thermoluminescent (TLD) Area Badges, room temp. self-annealing	24	At various points in Bldg 16A	Would be read in case of an event to develop iso-dose lines and estimate exposure to personnel
Personnel Monitoring: Hands, Clothes etc.	Eberline Instrument Co. Monitor Model RM-3C with AC-3 Alpha Probe 0-500 cpm	3	At clothes change area	
Film Badges with high level dectector strips	From commercial supplier such as R.S. Landauer & Co., Tracerlab or other Company approved		Worn on person	Read once each month. High level strips read in case of criticality event

7.24 Emergency Equipment--A cache of emergency equipment is maintained away from the working area in a structure located south of the 16A facility and near the south entrance in Building 2. The emergency station equipment **will include those items listed on the posted inventory sheet. They will include such items as:**

1) Scott Air-Pak (2)

2) First Aid Kit equipped for treating hemorrage, lacerations, burns, and shock.

3) Disposable suits, boots, head covers, and full-face respirators.

4) Decontamination equipment, detergents, and waste containers.

5) Portable Survey Meter, Victoreen Instrument Co. Model Thyac III.

6) Portable ion chamber 0-1000R, Eberline Instrument Model PIC-6A

7.25 Industrial Safety Equipment--In conjunction with the radiation safety program at the site, industrial health and safety of plant personnel is a continuing concern which is given considerable attention in all projects on the site. Some of the major equipment and facilities which are available to protect property and personnel include portable fire extinguishers, fire hoses and automatic sprinkler systems, a well equipped dispensary attended daily by a registered nurse, and a wide range of typical industrial safety equipment for individual and general use.

8.0 RADIATION SAFETY PROGRAMS AND PROCEDURES

A system of instructions is established whereby personnel at the facility are informed of policies, procedures and announcements relating to safety.

8.1 Programs and procedures are established as needed to minimize exposure to radiation. Steps are taken to familiarize all personnel who might be subjected to radiation exposure in the Nuclear Chemistry Facility with the programs and procedures. It is the philosophy of the Company to keep radiation exposures to the lowest practicable level in every instance.

8.2 Personnel Work Rules-- Work rules are established to minimize the hazards to personnel handling and processing radioactive materials. Examples of these rules are:

a) Protective clothing, as specified in paragraph 7.23 is furnished by the company and the type of clothing to be worn in a specific activity is based upon the potential for contamination from that activity.

b) Outer footwear worn in restricted areas is not to be worn into unrestricted areas.

c) Preparation, storage or eating of food is not permitted in process areas.

d) Hands are to be washed before eating.

e) Smoking is not permitted in radioactive material areas.

f) Safety goggles, safety glasses or other personal protection gear specified by signs or instructions for use in a

particular area must be worn at all times in that area.

g) Coveralls or laboratory coats shall be worn at all times in areas containing radioactive materials.

h) Tampering with film badges or other safety equipment will not be tolerated.

8.3 Radiation Limits in Controlled Work Areas--All areas at the Center where radioactive materials are stored or used are classified as radiation areas as defined in 10 CFR, Part 20.202. Design levels of penetrating radiation are well below the maximum level of exposure established in 10 CFR, paragraph 20.101. Penetrating radiation levels in occupied areas do not routinely exceed 1 mr/hr. It is not anticipated that any individual will receive a radiation exposure in excess of that specified in 10 CFR, Part 20.101 (a); but in the event an unusual condition should occur, under which an individual might receive greater exposure than is permitted by said sub-paragraph (a) then the limits specified in sub-paragraph (b) shall be met. Any area found to be a high radiation area as defined in paragraph 20.202 (b)(3) shall be roped off and labeled as a high radiation area. Entrance to a high radiation area shall be in accordance with the provisions of paragraph 20.203 (c)(2).

A dispensary is maintained at the Center for the immediate care of all minor injuries, such as minor burns, abrasions, bruises, etc. All visits to the dispensary by employees are recorded in a log book, and this information is transferred to the employee's permanent medical record.

All employees are instructed and advised about the chemical toxicity of various material used in the plant, chemical burns, minor injuries, and personal hygiene measures that should be followed to minimize the intake of all chemical substances that might be injurious to their health.

Close cooperation with the Health and Safety Department is maintained and all results of sampling for radioactivity and film badge readings are normally reviewed by the Medical Director.

8.6 Surveys--When working with radioactive materials, the work area is continuously sampled for airborne activity to determine that concentrations are at acceptable levels. Breathing zone areas, praticularly in locations where airborne activity is most likely to occur, are surveyed periodically with portable instruments. The sampler intake is located as near as possible to the worker's breathing zone. The maximum acceptable level of alpha activity in air, as represented by the survey sample, is 1×10^{-10} µCi/ml. In the event that this limit is exceeded, the operation will be discontinued and surveys made to determine the source of the activity. Steps will be taken to physically correct the abnormality. The continuous

air samplers are calibrated monthly with a flowmeter. The portable samplers are checked in the same way. Records will be maintained to show the check date and calibration data.

Contamination surveys will be performed in both restricted and unrestricted areas with sufficient frequency to assure that acceptable contamination levels are maintained. Particular emphasis will be placed upon surveying areas where the material is not in a closed system. These surveys will include frequent inspection of glove box gloves. The following contamination levels are to serve as action level guides above which the area will be decontaminated to levels below these guides and the cause will be investigated and, where reasonably possible, it will be corrected.

Area	Total Activity 1/	Smearable Activity 2/
Unrestricted	α 500 cpm $\beta-\gamma$ 200 cpm 3/	500 d/m 500 d/m
Restricted	α 10,000 cpm $\beta-\gamma$ 2.5 mrad/hr	5,000 d/m 5,000 d/m

NOTES:

1/ Values of alpha in counts per minute (cpm) are as read with a PAC-3G survey instrument, not more than one centimeter from the surface.

2/ Smearable alpha values are in disintegrations per minute (d/m) per 100 square centimeters.

3/ Beta-gamma values in counts per minute are as read with a Thyac III survey meter at one centimeter from the surface.

8.7 Any equipment or packages to be removed from the radioactive materials area will be surveyed and released by the Health and Safety representative. Radioactive contamination of all packages and equipment leaving the restricted area in Bldg. 16-A, Bldg. 20 and the enclosed yard shall be within the following limits:

1. The maximum amount of fixed alpha activity in disintegrations perminute per 100 square centimeters shall not exceed 25,000.
2. The average amount of fixed alpha activity in disintegrations per minute per 100 square centimeters shall not exceed 5,000.
3. The maximum amount of removable (capable of being removed by wiping the surface with a filter paper or soft absorbent paper) alpha activity in disintegrations per minute per 100 square centimeters shall not exceed 1,000.
   a) The maximum level at one centimeter from the most highly contaminated surface of a building or piece of equipment measured with an open-window beta-gamma survey meter through a tissue equivalent absorber of not more than seven milligrams per square centimeter shall not exceed one millirad per hour.

b) The average radiation level at one centimeter from the contaminated surface of the building or equipment measured in the same manner shall not exceed 0.2 millirad per hour.

8.8 When radioactive materials are in the plant, the survey instruments are calibrated per schedule as follows:

a) Instruments reading alpha are calibrated with sources furnished with the instrument or with standards prepared by the National Bureau of Standards or equivalent.

b) Instruments reading Beta-Gamma are calibrated with a cobalt 60 source.

c) The gamma alarm system is calibrated with a cobalt 60 source monthly. The system is checked weekly by visual inspection to assure that the fail-safe light is functioning.

8.9 Air sampling vacuum pumps are removed from service periodically and reconditioned. They are then recalibrated with a flowmeter and tagged with flow data and date of calibration.

8.10 Fire fighting equipment including hoses and extinguishers are checked periodically as required by the State and the insurance carrier.

8.12 Waste Disposal

The wastes generated in the facility are either liquid or solid. Procedures are established whereby:

a) Solid radioactive wastes are packaged in DOT approved packages for shipment to disposal.

b) Airborne solid radioactive wastes are removed by way of the ventilation system. Radioactive dust is evolved primarily in ventilated glove boxes. Each box or closed train has a prefilter which may either be disposed of as above when the pressure differential indicates that it is plugged, or it may be packaged and sent out for recovery of uranium values. All ventilation gases finally exhaust through filters designed to remove 99.9% of particulate 0.3 microns in size. All primary filters are of the fire-resistant type. When contaminated, these filters are shipped to disposal in DOT specification or special permit containers.

c) Liquid wastes include overheads from evaporation and various wash solutions. All solutions that could contain uranium are monitored, as described in Section II, and any solution that contains 2.4×10^{-5} microcuries of uranium per milliliter or less is released to the sewer. Solutions more concentrated in uranium are treated by distillation. During the infrequent occasions (estimated at not more than one week each year) when the boildown may become inoperative, solutions containing up to 3×10^{-5} microcuries per ml of activity, the limit set forth in 10 CFR paragraph 20.106, may be discharged to the sewer.

8.13 Reports and Records-- Individual exposure records obtained from the monthly reading of the film badges are maintained in the permanent records of each employee as are the results of individual clinical tests relating to radioactivity. An AEC Form 5 File is maintained for each employee. Quarterly entries will be posted. The activity in the work area as represented by air samples and smear tests are logged in graphic form to alert supervision to any undesirable trends, so that timely corrective action can be taken.

8.14 Industrial Safety Program-- The W. R. Grace & Company has long been involved in chemical operations involving toxic, flammable and corrosive materials. Over the years it has developed an excellent Industrial Safety Program. In addition, most of the employees at the Center who will be working in these facilities have had prior safety training in school and in industry. Each employee is instructed in safe practices, and the responsibility for working in a safe manner is placed upon him.

The benefits of training in operating safety and in good housekeeping practices carry over into the nuclear area.

Procedural controls are placed upon the storage and use of flammable, corrosive and toxic chemicals. Modern fire fighting and emergency equipment is available and fire prevention practices are followed in accordance with applicable codes.

8.15 Emergency Plans--The Washington Research Center has a comprehensive emergency procedure for the protection and evacuation of personnel in the event of a non-nuclear emergency such as an explosion or fire. To this procedure has been added the procedure to be followed which will minimize exposure to personnel and damage to property in the unlikely event of a radiation emergency.

8.16 The fire protection plan has three basic elements. The elements are:

a) An adequate alarm system that will alert everyone on the site that an emergency has occurred and that they are to evacuate.

b) An indoctrination program by which all personnel on the site will know what to do if the emergency alarm sounds.

c) A predesignated organization that has the training and equipment to cope with the emergency.

Arrangements to detect an emergency and initiate these procedures through the guard force on off-shifts is provided for in the procedures. Howard County has an emergency center which will furnish police, fire department, and ambulance service to area hospitals in the event of an emergency. The phone number of the Emergency Center is posted at each phone. Company vehicles are also available to transport injured personnel to the hospital. The guard also has a copy of the list of people to call in the event of a radiation emergency, and he is instructed to assist in keeping people away from either Building 16-A or Building 20, whichever is the source of the radiation emergency.

8.17 The radiation emergency plan and the procedures for coping with radiation emergency have been superimposed upon the emergency procedures already operative at the site. A copy of this procedure is furnished with this application. A radiation emergency is made known by means of the alarm system described in paragraph 12.12. In the event of a radiation alarm originating in Building 16-A, only the people in Building 16-A and the adjoining Building 16 evacuate and they go around Building 2 to the rendezvous room in the south side of the building. All others are cautioned to stay in their buildings and when those outside hear the radiation sirens, they are instructed to go inside the nearest building in a direction away from Building 16-A and Building 20. In this way, they can avoid being exposed to fallout. The closest building to 16-A is Building 2 and the nearest end of this building is 173 feet from the nearest wall of 16-A. Without considering the radiation attenuation afforded by the two sets of brick veneer walls, we find that a dose of 1000R one-meter from the source, when divided by the square of the distance, results in a dose of 348 mr at the nearest point in Building 2; therefore, in our judgement, the least overall exposure to personnel in other buildings will result if they stay inside until the extent of the emergency can be determined.

In the event of a radiation alarm in the storage area in Building 20, both Building 20 and Building 11, which is adjacent to it, evacuate to the Building 2 rendezvous area. All other inhabited structures are more than 175 feet away (See figure 3.2) and need not be evacuated until the extent of the emergency is determined.

The extent of radiation exposure to personnel is ascertained immediately by questioning personnel and with portable meters. Those who have received large exposures are evacuated to the hospital. The Health and Safety Supervisor will assist the physician in obtaining these data. The Medical Director is Dr. Mass, who will be working with Dr. M. J. Wizenberg and Dr. E Neptune of the University of Maryland Hospital. Drs. Wizenberg and Neptune are trained in the treatment of victims of radiation exposure.

An emergency equipment box containing portable detection and dose rate instruments, suits and airpacks, and a copy of the emergency procedure is maintained near the south entrance of Building 2.

8.18 It shall be the responsibility of the General Manager or his designate to notify the AEC of the accident. The Plant Manager or his designate will establish when personnel may return to the various work areas on the site and to coordinate decontamination during the recovery phase.

8.19 Radiation Safety Review--Any new facilities or any changes in facilities or operating procedures having to do with radioactive materials must be reviewed and approved by the Nuclear Safety Committee. The principal area of interest in this review other than nuclear criticality safety is to assure that adequate health protection facilities have been provided for safe operation. A decision by the Committee that operational safety is inadequate will not be overruled.

8.20 In the event of an emergency in the Building 16-A facility, and in the absence of the Plant Manager and the Production General Foreman, the senior man in Building 16-A will have responsibility for clearing the building, notifying the designated individuals and keeping people away from the facility until the emergency control organization can take charge. The Emergency Controller, or his alternate, has the responsibility for taking the actions necessary to cope with the emergency following the procedures in the WRC emergency plan.

8.20 The senior man who will be responsible in the event of an emergency will have the following minimum qualifications. He shall have a high school education or equivalent and at least one year of experience in nuclear processing. He shall have qualified to perform all of the operations being carried out and he shall have received training and understand the procedure he is to implement in the event of an emergency.

The emergency controller shall be a responsible member of management who is selected to perform the emergency controller function based upon demonstrated leadership and planning ability and his ability to make decisions under stress. He shall be familiar with all facets of the physical plant, the utilities and the emergency equipment at the site and its use. His work shall be at the site with a minimum of travel involved.

9.0 NUCLEAR CRITICALITY SAFETY

9.1 Summary of Program and Philosophy--The criticality safety program at the Center requires that each activity involving special nuclear material be analyzed for hazard potential, and that appropriate engineering and administrative safeguards be established to reduce to acceptable limits the potential for accidental criticality.

9.2 It is recognized that the only mishap in the operation that could instantly cause severe physical damage to personnel on site is from accidental criticality. It has been seen at other locations that such an event could be lethal to individuals in the immediate area. These events have also demonstrated that there is only moderate danger to personnel a short distance away, and very little danger to the public off-site in instances where the operation is relatively isolated as it is at the Center.

9.3 It is the philosophy of operation at the Center to rely upon engineered safety features in so far as is possible, and fortunately the type of operation performed permits extensive use of rigid equipment of safe geometry as the means of maintaining a subcritical condition. The process control group constantly checks the operation for strict adherence to the approved operating procedures including criticality controls. This is supplemented by frequent inspections by the Health and Safety specialist and at least once a month,when operating,by the Nuclear Safety Committee.

9.4 The limits selected to assure nuclear criticality safety in the storage, handling and processing of special nuclear material at the Center are listed in Table 9.4. The conditions applicable to these limitations are discussed in subsequent paragraphs. Safe limitations may be derived

TABLE 9.4

SPECIAL NUCLEAR MATERIAL LIMITATIONS

APPLICABLE TO THE OPERATIONS SUBJECT TO THIS LICENSE

Description of Material	Limiting Parameter	Operating Limit	Minimum Critical	Reference
1. Solution fully reflected	Mass of U-235	350 grams	820 grams(if reflected)	TID 7016 Rev. 1
2. High fired UO_2 full reflection, anhydrous	Mass of U-235	11 kg	>22.8 kg-130 kg (if reflected)	TID 7016 Rev. 1 Tables IV, I
3. Solution-Any U concentration full reflection	Cylinder Diameter	5 inch pipe	5.4 inches reflected	TID 7016 Rev. 1 Table 1 and App. A
4. Solution, maximum diameter-schedule 40 pipe with no reflectors immediately surrounding	Cylinder Diameter	6 inch pipe	>7 inches for 0.280 inch thick pipewall	TID 7016 Rev. 1 Fig 3, TID 7028, Fig 10
5. Cooling Jackets, maximum diameter schedule 40 pipe with no reflectors immediately surrounding	Diameter	6 inch pipe	>7 inches	(Same as #4)
6. Dense UO_2 up to 5 g/cc packed density (4.4 g/cc U-235), full reflection	Cylinder Diameter	4 inch pipe	>4.4 inches	TID 7028-Fig 10 and App. A
7. Dense UO_2, 7.7 g/cc packed density (6.7 g/cc U-235) full reflection	Cylinder Diameter	3.5 inch	>4.1 inches	TID 7028-Fig 10 and App. A
8. Spills or leaks onto floor (all solutions)	Slab thickness of catch tray	1½ inches	>2 inches reflected one side	TID 7028-Fig 11 and App. A
9. Waste Solutions	Neutron Poison in form of borosilicate glass raschig rings; filled containers in compliance with proposed ANS standard for such use	None for solutions	None for solutions	
10. Scrap solutions of known U-235 content	Concentration of U-235 in solution	5 g/1	12 g/1	TID 7016 Rev 1 Table 1

from values presented in technical publications such as TID 7016 Rev. 1, Nuclear Safety Guide, TID 7028, Critical Dimensions of Systems Containing U-235, Pu-239 and U-233, and others. Consideration has been given to higher enrichments, up to 97% U-235, which are contemplated. Their effect upon criticality parameters is examined in Appendix A.

9.5 Mass Limits--In the process itself there is no step where mass of U-235 in solution is the only criterion controlling nuclear criticality; however, this is the limit set for accumulations of materials in uncontrolled array such as samples and is the limit for the entire analytical area.

9.6 The high fired UO_2 in the process is devoid of hydrogenous material, hydrogen being the only effective moderator found in quantity in the entire process and therefore it is essentially unmoderated. The density of the UO_2 may approach theoretical for individual particles, but experience indicates a void fraction greater than 0.3 will be found resulting in an overall density in any accumulation of particles of less than 7.7 g/cc (10.9 g/cc x 0.7) and a U-235 density of less than 6.7 (7.7 g/cc x 0.88 U fraction). This places the critical mass for a fully reflected sphere of particles between 22.8 kg, the minimum critical mass for uranium metal at full density, and about 130 kg the critical mass for unmoderated salts. Absorption of water into this material, which is not hydroscopic, is precluded by keeping the material in sealed containers except when operations are being performed upon the material and this is done only in a glove box enclosure maintained under inert atmosphere. There are no sprinklers in the glove box area.

9.7 Geometry Limits--A series of cylinder diameter limitations are used in the process and in storage to assure safety under any foreseeable conditions. As shown in Figure 9.4--Item 3, the process solutions, many of which are near optimum concentration, are restricted to 5 inch cylindrical containers. None of the containers have close fitting reflectors and all have been spaced from walls to minimize neutron reflection from this source. In the case of solutions that normally contain from a trace of fissionable material to perhaps 20 g/l and in the case of cooling jackets, the diameter is increased and then only to that of 6 inch schedule pipe. No reflectors immediately surround these containers, the closest reflector being one wall at least 15 inches distance, thus assuring safety at any fissionable material solution concentration for the reflection available. The close fitting reflection is equivalent to about 0.280 inches of water, or less, which is the steel or PVC wall thickness, the chloride content of the PVC acting as a neutron absorber. The combination of normally low fissile solution concentration and less than nominal reflection provides an additional safety factor in the use of 6 inch schedule 40 pipe.

9.8 Once the oxide has been taken out of solution as solid particles, the density increases. Unfired particles could be as dense as 5 g/cc; although they more nearly average 3 g/cc. The container diameter used for these particles has been established at 4 inches. Conservatism is found in that the containers are of finite length of less than 3 feet and in none of the locations are they fully reflected. The densified particles are confined to containers measuring no more than 3.5 inches in diameter. Although the particles are only partially reflected in any location, safety is assured in the event of full reflection.

9.9 Under each grouping of equipment containing fissile material in solution is a chemically resistant tray which is sized at $1\frac{1}{2}$ inches depth to hold the contents of the largest single container or group of containers that could drain or siphon out together. These are compared individually in Table 9.9. Assuming full reflection from the floor and practically no reflection from the open top, this depth which would be subcritical for a fully reflected slab has ample safety.

9.10 Neutron Poisons--Poisons are not used in the process. Fixed poison in the form of borosilicate raschig rings which contain 12.5% natural B_2O_3 and are sized to uniformly occupy 23% of the tank volume are used in the waste solution collection tanks 10-42A and 42B all in compliance with proposed ANS standard entitled "Use of Borosilicate Glass Raschig Rings as a Fixed Neutron Absorber in Solutions of Fissile Material"--November 1965. The waste entering these tanks normally contain only traces of special nuclear material. The fixed poison in these tanks provide back-up protection against a failure elsewhere in the system. When the tanks are full, the solution is analyzed for special nuclear material. It is discarded to the sewer if found to be 2.4×10^{-5} μCi/ml or below, otherwise it is retained for recovery of the uranium via the boildown system. These tanks are maintained at a pH of four or less by acid additions to prevent uranium precipitation.

9.11 Concentration Limits--Throughout the process the solutions are, for the most part, confined such that they are safe by reason of features other than the concentration of fissionable material. Solutions such as waste solutions for scrap recovery which contain, by analysis, 5 g/l or less U^{235} may be safely stored in containers of any size. Precipitants are carefully excluded to avoid concentration by settling. The containers

COMPARISON OF SOLUTION TANK CAPACITY WITH CATCH TRAY CAPACITY

Fissile Solution Tank No.	Description	Tank Volume cubic inches	Catch Tray Size inches	Tray Volume cubic inches
13-1	Dissolver	377	48 x 48 x 1½	3450
	New Dissolver	942	"	"
10-3A 10-3B 10-3C	Dissolver Product Hold Tanks	3140 each	84 x 108 x 1½	13608
10-4 10-5 10-7	Dialysis Tank Aerator Anolyte Tank	3375 Largest	84 x 108 x 1½	13608
10-10A 10-10B 10-11	Sol Storage Tank Sol Storage Tank Column Feed Tank	3140 each	84 x 108 x 1½	13608
10-37A 10-37B 10-39	Boildown Feed Boildown Feed Boildown	3391 Largest	84 x 108 x 1½	13608

of these solutions are stored above the freezing temperature of the solution to preclude the possibility of selective crystallization of the uranium compound.

9.12 Neutron Reflection--Process vessels containing solutions of fissile materials have been located a foot or more from building walls which lessens neutron reflection from the wall, and for operating convenience. This process system, as designed, is safe in that the five inch vessels are safe fully reflected, see item 3 Fig. 9.4 and the six inch vessels that could contain fissile material are conveniently located 3 1/2 feet or more distance from the nearest wall and, therefore, have reduced reflection so that item 4 of Fig. 9.4 applies.

9.13 Interaction Between Units-- As seen in Table 9.4 the container diameters selected are subcritical with full water reflection. It was convenient from a process standpoint to place groups of columns of fissile material in single planes so that an individual column will see a relatively small area of fissile material, and this was done.

The columns of solution are 27 inches apart on center and 36 inches or more separates groups. It has been computed that a six inch air gap between an infinitely long cylinder of U (93) solution at 500 g U-235 per liter and an annulus of the same solution would be subcritical for a cylinder diameter of less than 7 inches (LA-3366 Fig. 7). The smaller diameters and relatively large spacings used in the design of this plant are found by inspection to afford a wide margin of safety. Interaction by the solid angle method was also determined for the various groups of containers and systems in the area. The results of these calculations are given in Table 9.13. Details are found in Appendix B.

TABLE 9.13

INTERACTION BETWEEN UNITS IN REACTING ARRAYS

Description	Calculated 1/ Individual Unit K_{eff}	Permissable Total Solid Angle In Steradians	Actual Total 1/ Solid Angle of Array Steradians
Particle Storage Array	0.54	3.7	1.4
Solution Storage in Tanks 10-3A, 3B, 3C, 10-4, 10-5, 10-7, 10-10A, 10-B and 10-11	0.74	1.5	0.82
Particle Columns	Very Small	Large	0.36
Boildown Feed Tanks Boildown Evaporator	Small	Large	0.5

Note 1/ Calculations are included in Appendix B.

9.14 Design Integrity of Structures

(a) Whenever criticality control is directly dependent on the integrity of a structure used to retain the geometric form of a special nuclear material accumulation or the spacing within a storage array, the structure will be designed with an adequate strength factor to assure against failure under foreseeable loads or accident conditions. Materials of construction will be selected to resist fire and the degree to which any corrosive environment might affect nuclear safety will be considered and corrosion-resistant materials or coatings applied as necessary.

(b) Whenever criticality control is directly dependent on the integrity of a neutron isolating structure, the structure will be designed to assure against loss of integrity through foreseeable accident conditions such as fire, impact, melting, corrosion or leakage of materials.

9.15 Miscellaneous Containers

Dense particles are collected only in the 3½ inch diameter containers and none larger are permitted in the particle area. Spills of solutions are cleaned up with a sponge and the solution is collected in a five inch diameter rigid bottle. Only this size container or smaller will be permitted in the operating area. Each five inch diameter bottle will be labeled "UNSAFE FOR USE WITH DENSE PARTICLES".

SAFETY ANALYSIS

General--In this section each process step is discussed relative to (a) Nuclear Safety, (b) Radiation Safety, and (c) Safeguarding of Special Nuclear Material.

Dissolution--A quantity of uranium oxide powder or recycle particles, not to exceed 3 kg of U-235 is first weighed out in a glove box, then it is poured into the top of the dissolver. An inorganic acid is introduced to the dissolver, the system is agitated and heated to effect solution, or in the alternate system it is circulated through a heat exchanger to effect agitation and to control temperature. After solution is completed, the system is cooled and the uranium containing solution is drawn off through a small filter to storage.

(a) Nuclear Safety.

One container of dry oxide from the shipping package or one standard 3.5 inch diameter canister of particles for recycle and the 3.5 inch diameter dissolver charging bottle are permitted in the glove box weigh station at a time. The combined volume of the containers does not exceed 3.0 liters, the safe fully reflected volume for uranium oxide from Figure 2,TID 7016 Rev. 1. The material is moved in this closed container to the dissolver hood, posing no nuclear safety problem.

The dissolver is four inches in diameter by 30 inches long and it is therefore of safe geometry for the reasons given in paragraph 9.7. The dissolver jacket is a six inch schedule 40 pipe which normally contains no uranium but it is safe in the event solution leaked into it because it meets the conditions of item 5 in Figure 9.4. The alternate dissolver is 5 inches in diameter and it has bare walls. It is safe for the reasons given in paragraph 9.7. As back-up against loss of fissile material to unrestricted areas the cooling and heating system is a closed system with a three inch diameter tube and shell heat exchanger 02-1 spaced 27 inches on center from the dissolver. The dissolver operates at atmospheric

pressure, whereas the cooling system is maintained under 15 psig so that a leak would result in a flow into the reactor. The heating-cooling system pressure and the liquid level in its expansion tank are both read and recorded before uranium is introduced into the dissolver for each dissolution. If either are found to be low, they are corrected. The dissolver is then inspected and a determination is made that it does not leak before proceeding with dissolution. With the five inch dissolvers heat is transferred through a 2 inch diameter by 24 inch long vertical heat exchanger located above filter 06-2B which is 30 inches from the dissolver. The solution filter 06-2A or 2B measures four inches in diamter and is therefore safe. The uranium solution is stored in three five inch PVC pipe columns which are bare and which are located in one plane on 27 inch centers. To protect against distortion of these tanks, should the dissolver solution inadvertently be introduced into a tank while still hot, a steel sleeve surrounds each of the three tanks.

(b) Radiation Safety.

The system is a closed solution system except for the introduction of urania powder into the top of the dissolver. To minimize dusting in transferring the powder from the bottle to the dissolver, the bottle is conical in shape at one end and this cone is inserted into the small part in the top of the dissolver is inside a small box with a sliding window front. This window is normally closed. The face velocity with the window open exceeds 100 fpm. This enclosure is vented through a small absolute filter into the process ventilation system.

(c) Safeguards.

Since the uranium oxide loses its identity at this point of introduction into the system, the quantities introduced are carefully measured and a record is made of the weights.

10.3 Dialysis--Approximately 40 liter batches of dissolver solution are converted to a sol by circulating the batch through a dialysis cell. The finished sol is transferred to the sol storage tanks. The non-uranium bearing products of dialysis are taken into the anolyte solution. This waste solution is treated by way of the boildown system. The product gases are scrubbed with caustic in tank 10-8. This solution is barren in uranium and when spent, it is discarded by way of the acid waste tank system 10-42A or 42B.

(a) Nuclear Safety.

The PVC dialysis cell has a total volume of 1.5 liters and is therefore of safe capacity even if the membrane were to rupture and a significant concentration of uranium were to enter the anolyte side. The walls are relatively thick at nearly two inches which when added to the six feet of distance to the nearest unit avoids significant interaction between the dialysis cell and other vessels.

All of the tanks in this system are five inch schedule 40 pipe and all except 10-8, the scrub tank, which tank does not contain uranium, are in the same plane as the dissolver product tanks and these tanks are spaced 27 inches on center. This arrangement is shown to be safe for interaction in paragraph 9.13. In the event of a membrane failure

the anolyte solution could contain sufficient fissile material to require geometric restriction. This solution, in any event, is pumped to the boildown feed tanks 10-37A or 37B. These tanks are individually safe by reason of item 4 of Table 9.4. Both the anolyte and catholyte systems pump solution into the dialysis cell and overflow back to the batch feed tanks. These overflows are at a lower elevation than is the vent line to the scrub tank 10-8 and with a broken membrane they would become one equal pressure system, consequently solution cannot reach the vent which feeds the scrubber tank 10-8. In any event, this five inch schedule 40 pipe tank would be safe with fissile material in it by reason of Table 9.4--item 3.

(b) Radiation Safety.

This is a closed system and it poses no undue radiation problems; however, employees are instructed not to remain in close proximity to these units except as needed to perform the required operations.

(c) Safeguards.

No special nuclear material measurements are made at this point in the process and the only special nuclear material leaving the process proper is that which goes to the boildown system where it is concentrated and the uranium content is measured at that point.

10.4 Particle Formation--A batch of sol is pumped to the particle column feed tank. The sol in this tank is continuously circulated through a four inch diameter by eighteen inch long constant overflow tank which feeds the top of the column through proportioning pumps. The sol flows down the column and is dehydrated by an imiscible drying agent which flows

the five inch diameter point by the above reference. The volume of the two inch to five inch section is 1.9 liters and would contain 3.8 kg of particles. The procedure requires that the particles be removed in approximate 400 g batches, thus there is a wide margin of safety to reach 3.8 kg.

The drying agent, as it leaves the column, normally contains no uranium since it is insoluble in this solution; however, a malformed particle connected to a gas bubble does carry out with the drying agent on occasion. Any such material is removed by four inch diameter cartridge filters which are used alternately. The agent is then scrubbed with water in a six inch pyrex glass column to remove chemicals which is followed by a six inch glass column drying agent-water separator then to redrying.

It is to be noted that there is no mechanism by which a gross quantity of fissile material can be carried to the four inch filter. Experience indicates that it takes months to collect a few grams of material. When the filters are replaced, a few particles may carry through and settle in the bottom of the water scrubber column. These particles are drained into a small beaker and transferred to the scrap container. The mechanisms that assure nuclear safety in the drying agent recycle system are:

1) Greater density and immiscibility of fissile particles prevents gross carry-over with the drying agent.

2) A filter effectively removes the few grams that occasionally do carry-over.

3) In the event of filter failure particles can be readily observed in the bottom of the glass water scrub column from which they are drained whenever they appear.

(b) Radiation Safety.

This system is a completely closed liquid system which poses no new problems.

(c) Safeguards.

Material leaves the system as product from the column which is handled as discussed in the next paragraph and a very small quantity may find its way to the cartridge filter which material is eventually accounted for as scrap.

10.5 Particle Washing and Drying--The small beaker of particles taken from the column is transferred to one of the four inch wash containers which is drained of drying agent, or to the smaller fixed washer tube in which the drying agent is displaced by an aqueous solution. The drying agent is piped by way of a cartridge filter to the redrying system. The particles are rinsed with an aqueous solution and placed in a drying tray. These trays, not to exceed one inch in depth, are placed in the vacuum furnace and the particles are dried. Alternatively the three inch diameter tube furnace may be used for this operation. The aqueous wash passes through the particulate filter and to the waste water collection tanks.

(a) Nuclear Safety.

The particles are maintained in safe geometry containers of four inch diameter, or in safe volume slabs. The maximum volume in one location is 4.4 liters of total tray capacity in the furnace. No other trays are permitted in the area and the furnace opening is fitted to accommodate only these trays. The material is not densified appreciably in the dryer but the moderator is removed thus decreasing reactivity.

The maximum particle density at this point is 35% of theory giving a maximum random density in the tray of 2.1 g/cc U-235. The maximum charge

to the oven based upon trays filled 3/4 inch full on the average would be approximately 7 kg of U-235. The trays actually form two separate slabs each having a capacity of 2.2 liters. Optimum moderation is assumed as is full reflection because the furnace is water cooled and flooding although unlikely is possible. The safe volume for a sphere with optimum water moderation after adjustment for increased enrichment is 4.5 liters from Fig. 2 TID 7016 Revision 1. The volume of all the fissile material containers in the slab array encountered here is less than 4.5 liters, and the array is less reactive due to lower neutron economy from the much greater surface area present. A metal baffle is interposed between the cooling coil and the particle trays so that in the event the cooling coil should leak, the water spray will not displace the particles from the trays. In addition, administrative procedures will provide periodic observation through the window and vacuum gauge to detect the presence of water. If water is detected, the cooling coil is immediately shut off and the furnace unloaded. The alternate fixed, washing units are in two vertical planes with 28 inches spacing between the two planes. The first pipe tank in each unit is located 24 inches from the particle column it serves. The remaining tanks are in a straight line 18 inches on center extending away from the particle column. These pipe tanks have diameters of three inches or less except for the 4-inch diameter solution reservoir at the end. The pipe tanks are bare and extend into the open area. The heights of the tanks are in every instance less than the 54-inch height of the storage rack described in paragraph 10.9. All but one of the units are of smaller diameter, they are on the same center to center spacing and the distance between the two planes is greater by 4 inches than that of the particle storage area. It can be seen by inspection that the reactivity of this

array of small vessels is less than that calculated for the storage rack which was shown to be conservatively safe. Only one particle transfer vessel of four-inch diameter or less is permitted at either of the two systems at one time. The filter on the wash solution outlet prevents inadvertent loss should a hole develop in the collector screen. The wash water is collected in the fixed six inch glass columns to permit drying agent separation and then it is routed to poisoned tanks 10-42A or 42B.

(b) Radiation Safety.

This is the only operation in which fissile material is routinely handled in the open and with small quantities of wet material no dust problem is anticipated, but the operation will be carefully monitored. After the particles are dried, covers are placed upon the trays and they are immediately transferred to the closed system glove box.

(c) Safeguards.

Essentially all material remains together at this point and it is not removed from the process.

10.6 Sintering--The dried particles are placed in the three inch diameter sintering furnace in quantities of about three kilograms. They are heated under reducing atmosphere and densification takes place. They are cooled and removed from the furnace in the standard canister and transferred to an inert atmosphere glove box where they are weighed.

(a) Nuclear Safety.

One three kilogram batch at a time is processed through this operation. Safety is maintained by the geometry of the equipment which is less than 3½ inches in diameter, as discussed in paragraph 9.8.

(b) Radiation Safety.

The charge is placed in a loading device and sealed in the glove box. The transfer device is attached to the furnace for loading so the particles are not exposed and airborne activity from dusting is avoided. The use of argon purging with H_2 prevents an explosive mixture in the furnace. The off-gases pass through a clean gas dust filter then a primary high efficiency filter and into the ventilation duct. The normal air flow through the duct dilutes the H_2 to a factor of 100 below the minimum explosive limit of 4% in air and no explosion hazard exists.

(c) Safeguards.

At this point each sintered batch is carefully weighed and sampled for uranium analysis and a material balance is maintained accross the wet end of the process by comparing these measurements to the input measurements.

10.7 Dry-end Processing--A number of physical operations are performed upon the unmoderated spheres. These operations are performed in glove boxes which are ventilated as discussed in Section 12. No liquid or other moderating material is permitted in these glove boxes. Good physical specimens are coated in an essentially unmoderated system in that the only moderator being one atmosphere or less of H_2 gas. All transfers and stoarage is done in 3 1/2 inch diameter containers and the coating equipment does not exceed this diameter.

(a) Nuclear Safety.

Hydrogenous materials are excluded from the glove boxes permitting safe use of the unmoderated limits for sintered material set forth in Table 9.4 item 2, except in the weigh box where the material from the vacuum dryer is limited to three standard canisters with a combined volume of 3.3 liters. The vacuum dried material, unlike the sintered material, may contain a small percentage of chemically combined water, however, it enters only the weigh box and no other fissile material is permitted in that box at the same time. The balance of the glove boxes in this train is limited to 11 kg U-235 total and the containers of fissile material need not be in restricted array in the glove boxes and this applies to the finished product train also.

(b) Radiation Safety.

Transfers are made only in closed canisters and furnace loading is performed using the sealed system loading device. All other operations are performed in closed system glove boxes which are exhausted through high efficiency filters which are discussed in Section 12; therefore, there is no unusual radiation hazard in these operations.

(c) Safeguards.

Both scrap and product are carefully weighed before they leave the system and records are maintained of the quantities of each, and a material balance on these operations is kept.

10.8 <u>Finished Material Washing and Drying</u>--The coated particles are exposed to moderator once more when they are washed with inorganic acid, rinsed with water and then rinsed in alcohol. This operation is performed in a four inch diameter container of 2.5 liter capacity. Normally the coating is impervious to this treatment, however, the wash solutions are collected and analyzed to assure that this is indeed the case before they are discarded. Finally the product is placed in a tray and dried under vacuum at elevated temperature.

(a) Nuclear Safety.

A single container of this material is processed at one time and it holds a subcritical volume. The particles are dried in a slab of no greater volume. The washing acid solutions are collected in a three inch cylinder which is visually checked for particles. If it contains none, it is carried to the boildown feed pump and pumped by hose into the boildown feed system. It could only contain particles if the wash unit filter broke in which case, the unit would be replaced and the batch would be reworked. The alcohol and water washes which will not dissolve uranium and; therefore, cannot contain more than trace quantities are drained through a small cartridge filter which removes particulate and into a five inch pipe tank located beneath the glove box. This solution is drained into a five inch diameter container and carried directly to process waste tank 10-42A or 42B. Full bottles are not permitted to collect.

(b) Radiation Safety.

This operation is performed in a glove box as discussed in Section 12 and no unusual radiation problem exists.

(c) Safeguards.

Only possible loss of special nuclear material in this operation is in washing and it is insignificant; however wash solutions are checked before discarding to ascertain that significant quantities of uranium are not being discarded.

10.9 Particle Storage--There are a number of stages in the process in which particles are stored to await further processing. These particles are stored in the standard canister which measures 3.5 inches outside diameter by nine inches in length. The racks have locators to center the canisters on their supports into a vertical column of up to six canisters. In one case six of these racks are in line, which places the columns of canisters on 18 inch centers, in one plane. An identical plane is parallel to this plane with two feet between planes. A glove box is located at the end of the rack and it may contain the contents of one storage canister at a minimum distance of two feet from the nearest column of canisters in storage. If all 200 kg of U-235 authorized were in this storage rack, it would completely fill the rack with a U-235 density of 0.08 g/cc.

(a) Nuclear Safety.

These units are individually safe by the criteria of Item 7 in Table 9.4. The lack of reflectors provides decreased reactivity. An analysis of the interaction within the array shows the array to be safe (Table 9.13).

(b) Radiation Safety.

The canisters are covered with a sealing cover when in storage posing no dusting problems.

(c) Safeguards.

Each canister of material in the storage rack is carded to identify it.

10.10 Drying Agent Safety--Considerable care has been exercised in the design of the system for washing and drying the drying agent to minimize the fire hazard associated with the use of this flammable hydrocarbon. It has a flashpoint higher than that of kerosene. It has a very low vapor pressure at the temperature of operation. The ventilation exhausts at the bottom of the particle columns to remove any fumes. The drying agent is redried in the still room. This room is separated from the operating area by a concrete-block wall. It is provided with a blow-out wall to the outside as a safety feature and it is equipped with explosion proof electrical equipment and fixtures. It and the particle column area are the only ones sprinklered. As noted in section 12, the air is replaced 12 times an hour in this room. Thus it is apparent that any hazard associated with the use of this flammable liquid has been handled in a manner representing good industrial practice.

10.11 Sumps--There are two sumps in the 16A facility. One is in the floor of the change room and the showers, wash basins, fountains and laboratory sinks drain to this sump. Solutions are pumped from this sump to the shower waste holding tanks discussed in paragraph 11.2. The other sump is in the still room and provides a collection point for solution leaks in the still room.

(a) Nuclear Safety.

Neither of these sumps are accessable to any solution containing more than trace quantities of fissile material under any circumstances; consequently,

they need not be of restricted geometry. For any fissile solids to collect in the shower sump these solids would have to be dumped in the laboratory sink contrary to procedure. Solid fissile material could collect in the still room sump only by spillage when removing a filter or draining particles from the water scrub column in the unlikely event of filter failure. To avoid the long term possibility of fissile solids buildup, the solids in the bottom of the sumps are sampled each six months and if uranium is found the sumps are cleaned out.

(b) Radiation Safety.

No radiation problem is associated with the sumps.

(c) Safeguards.

Significant quantities of fissile material are not likely to be found in these sumps. This sump will be checked for uranium content if a spillage occurs but not less than once every six months.

10.12 Large Tanks--In addition to the tanks previously discussed, there are six large tanks associated with drying agent or chemical supply which are not of restricted geometry because fissile material cannot reach them. These tanks are as follows:

Tank Number	Description
10-21	Dried Solvent receiver
10-22	Spent Solvent Tank
10-23	Clean Solvent Storage Tank
10-28	Aqueous Wash Solution Storage Tank
10-29A,29B	Wash Solution Feed Tanks

(a) Nuclear Safety.

The drying agent passes through the washing system described in paragraph 10.4 before it can reach tanks 10-21 or 10-22, and tank 10-23 contains only new drying agent without provision for recycle; therefore, no fissile material can reach these tanks. The aqueous solution tanks 10-28, 10-29A and 29B are all located at a higher elevation than the systems they feed, so that siphoning cannot occur and they are not equipped for recirculating solution; therefore, no fissile material can reach these tanks either.

(b) Radiation Safety.

No radicactive material involved.

(c) Safeguards.

No fissile material involved.

11.0 WASTE TREATMENT AND DISPOSAL

11.1 Scrap Recovery--The scrap generated in the facility consists of liquids; solids consisting principally of non-specification products; and contaminated clothing and materials. Non-specification product may in some instances be recycled directly to dissolution. All other solid scrap will be packaged in DOT specification 6L containers and shipped to a licensee qualified to receive and recover the uranium in the scrap. The liquid wastes will also be sent out for recovery. These solutions are normally concentrated by distillation in the boildown system and they are packaged directly from the boildown into DOT special permit packages specifically approved for highly enriched uranium solutions under this license. Arrays of packages will be limited to the number permitted in shipment unless otherwise specifically approved by amendment to this license.

If after dual analysis a well mixed waste solution such as that collected from the boildown or that from tank 10-42A or 42B, is found to contain 5g U-235 per liter or less and it is in a form that will not precipitate it may be packaged in DOT specification drums without nuclear criticality limitations as to size or arrangement.

11.2 All liquid process wastes that are expected under normal circumstances to contain in excess of 3×10^{-5} microcuries of U/ml are routed directly to the boildown feed tanks 10-37A or 37B. All other squeous process wastes go directly to poisoned tanks 10-42A or 42B.

The showers, wash basins, fountains, clothes washing machine, and the laboratory sinks drain to the shower room sump from which they are pumped up to one of the shower waste holding tanks. The laboratory sinks are used for washing apparatus and hand washing. All analytical

solutions that may contain uranium are poured into five inch diameter bottles and the container is rinsed into the bottle thus no detectable uranium will normally be found in this solution.

When tank 10-42A, 42B or either of the shower waste holding tanks become full, flow is switched to the alternate. The solution in the full tank is cycled by pump for thirty minutes to mix the solution thoroughly and a sample is taken. If the uranium content of the sample exceeds 2.4×10^{-5} microcuries of U/ml, the solution is routed to 10-37A or 37B for further treatment in the boildown. Solutions containing 2.4×10^{-5} microcuries of U per ml or less are discharged directly to the sewer without further treatment.

Under normal conditions the distillate from the boildown contains no uranium and it will be discharged directly to the sewer unless the activity exceeds 2.4×10^{-5} microcuries per ml. During the infrequent occasions (estimated at not more than one week each year) when the boildown may become inoperative, solutions containing up to 3×10^{-5} microcuries per ml of activity, the limit set forth in 10 CFR paragraph 20.106,may be discharged to the sewer.

11.3 The plant sanitary sewer and chemical sewer systems are treated separately to meet the general and specific water quality standards of the State of Maryland for the Middle Patuxent River into which they are finally released. They are combined prior to release to a creek on the property. Combined workday flow will

average 4×10^5 liters, dropping off to 2.4×10^5 per day on week-ends. The 10^3 liters of process liquids to be discarded daily are thus subjected to dilution on the order of 100 or more prior to release.

11.4 In the course of the operation some disposable solid materials will undoubtedly become contaminated. For the most part these will consist of clothing, gloves and the like which will contain only trace quantities of fissile materials and which may be disposed of by burial at a licensed burial facility.

Material that could contain appreciable quantities of fissile materials such as filters will be monitored for activity. If the presence of a significant quantity of fissile material is indicated, the item will be packaged in DOT specification containers for shipment to scrap recovery. None of these materials pose a fire hazard.

11.5 The bottoms from the boildown are collected in a four inch diameter container and the solution is pumped into 5 inch pipe tank 10-41A or 10-41B. These tanks which were six inch diameter have been replaced with bare five inch diameter PVC tanks. When one of these is full of concentrated uranium solution, it is circulated until it is homogeneous then it is sampled and analyzed for uranium. It is then drained into the shipping container bottles. These bottles are loaded while in the shipping package or storage cage to assure safe spacing from other containers and tanks. The package is moved to the pad outside of the building and then to Building 20 for storage, again the number of packages permitted

in any array shall be as approved for transport or as permitted by amendment to this license. An interim storage area for 4 3/8 inch inside diameter polyethylene bottles up to 11 liters capacity is located just inside the west door leading to the pad. It will contain up to 30 fifty-five gallon drums not stacked with one polyethylene bottle held by a metal frame in the center of each drum. The interim storage drum birdcages have holes in the sides at the bottom so solution cannot accumulate to an unsafe depth in the unlikely event that a bottle should leak. The array is located eight feet from the nearest column containing fissile material which is the boildown system. Sixtyeight packages with this spacing are safe in one vehicle for transport under SP-5061 revised. From Figure 76 of TID 7028 and taking into consideration the increased enrichment under this license which is offset by the 7% smaller diameter of the bottles used here it is apparent that the number of containers to reach criticality with 48 cm edge to edge spacing as is provided here is many times the quantity stored.

12.0 AUXILIARY SYSTEMS

12.1 Ventilation System--The ventilation system for the Building 16A Facility consists of a combination of blowers, filters, ducts, dampers and other control devices which work together to provide a clean and conditioned air supply to all areas wherein radioactive materials are handled, and to exhaust the air from these areas thru filtering devices which will assure that any release of airborne radioactivity downstream from these filters is within the acceptable release limits of 10 CFR 20.

12.2 The basic design guidelines are similar to those covering the ventilation of radioactive operations as set forth in the Manual of Recommended Practice for Industrial Ventilation, published by the Committee on Industrial Ventilation of the American Conference of Governmental Industrial Hygienists.

12.3 The flow of air is designed to be never from areas of higher contamination toward areas of lower contamination; all building exhaust air is directed to the filtered exhaust system for final discharge above the building roof. The areas of lesser contamination, such as the working areas of the rooms, are always maintained at levels which allow personnel occupancy.

12.4 The fresh supply air to all working areas of the radioactive facility is 62% fresh outside air and the balance is recycled cooled or heated room air. This air is filtered and tempered by heating or cooling before being discharged to the work spaces. The amounts

of supply air discharged to these spaces is sufficient to maintain any level of radioactivity present well below the acceptable maximum allowable of 10 CFR 20, and respiratory protective equipment for the personnel will not be required for normal operations. In the event of unexpected air contamination above acceptable work levels, authorization is requested to use respiratory protective equipment pursuant to 10CFR 20.103(c)(3). The fissile material processed in this facility is not likely to cause contamination that cannot be cleaned up when wearing the usual work clothing plus respiratory protective equipment if air-borne activity is associated with the activity. Health and Safety will rope off any contaminated areas exceeding the action levels defined in Section 8 thereof, and they will specify the protective clothing and respiratory devices required to enter the area. Health and Safety will be called in and monitor the cleanup of any major spill. Contaminated materials, if solutions, are collected in five inch bottles and transferred to the waste solution system, or directly to the five inch DOT Special Permit shipping container for shipment to recovery if the solution contains appreciable uranium. Other contaminated materials are packaged directly in DOT specification drums for shipment to waste burial.

12.5 The air from the working areas is exhausted at a rate which maintains these areas at a slightly lower pressure than the surrounding areas and assures that any air leakage is inward toward

the areas of potential contamination and away from the non-contaminated areas. All of the exhaust air is passed through a bank of "absolute"type final filters which are greater than 99.9% efficient for 0.3 micron diameter particles. The filters are equipped with pressure drop measuring devices to indicate the amount of resistance developed due to particulate loading; the filters are factory tested to 10" w.g. pressure, and when the in-service pressure drop reaches 50% of the test value, the filters are discarded and replaced with new ones.

12.6 Within the work area, the radioactive material is processed in completely contained handling systems. In the solution form it is contained by closed tanks connected by a piping system. This system is kept and operated in a leak-tight condition; should any leaks inadvertently occur, chemically resistant floor pans are placed under all pieces of equipment so the liquid can be quickly recovered and not spread into difficult locations for retrieval; the fumes or vapors from any such occasional leakages are considered to be slight and they will be rapidly swept away by the copious quantities of fresh ventilation air supplied.

12.7 In the solids form, the radioactive materials are processed in a completely contained system of high-integrity glove boxes which are tested for leak-tightness before being placed in service. Whenever the materials are withdrawn from the glove boxes for storage or transfer, they are first placed in a small tightly sealed stainless steel flanged container or a screw top polyethylene bottle. These containers are wiped clean and passed from the glove boxes through two-door airlock entrances which effectively prevent the glove box atmosphere from contaminating the room atmosphere during the transfer operation. There are two hoods in the area with sliding doors, one at dissolver loading and one at needle cleaning. These have a face velocity of 100 feet per minute which velocity is checked at regular intervals under contract to an organization which tests and certifies the air flow to all hoods on site.

12.8 The glove boxes are separated into two separate systems. One glove box system operates with an air atmosphere in the interior. These boxes are equipped with a separate filtering system of their own. The boxes are operated at a negative pressure approximately 0.5" to 1.0" w.g. below the surrounding room pressure. Room air is drawn into the glove boxes through an inlet which is equipped with a high efficiency absolute type filter, at a rate sufficient to carry away any heat or fumes generated within the enclosure. In the unlikely event that the pressure within the boxes should exceed that of the surrounding area, the backflow air will be filtered as it leaves the box and radioactive particulate matter greater than 0.3 micron diameter will be trapped and retained within the filter. The exhaust air from the glove box system is withdrawn from the boxes through an outlet which is also equipped with a separate high efficiency absolute type filter. This filter prevents radioactive particulate matter greater than 0.3 micron diameter from escaping the confines of the glove box system. There are four groups of these air atmosphere glove boxes, each group consisting of either multiple glove boxes connected together, or a single individual box. Each group is equipped with an absolute type filter on the ventilation air inlet and outlet in the manner described.

12.9 The amount of negative pressure on each box line is regulated by means of an adjustable damper valve in each exhaust outlet. This valve adjustment can be set and locked in any desired position. The main ventilation exhaust system operates under a negative pressure of several inches w.g., and provides the source of negative pressure under which the glove boxes operate. The filtered exhaust ventilation air from the glove box system is directed into this main ventilation exhaust system upstream from the final filters. It then passes thru these final filters, and is, therefore, filtered twice by high efficiency filters before being released to the environment.

12.10 The other glove box material handling system operates under an inert gas atmosphere. This system consists of multiple glove boxes connected together and having double door gaslock type entrance ports. Means for evacuating the gaslock ports during transfer operations are provided. The glove boxes are tested for leak-tightness upon assembly, and the inert gas atmosphere within the boxes is maintained at a slightly positive pressure of about 0.1" to 0.2" w.g. with respect to the surrounding room pressure. This pressure is maintained by an automatic pressure regulating valve which admits the inert gas in the quantity required and maintains the pressure at a constant setting. The inert gas inlet flow rate is determined by the withdrawal rate, which is regulated by means of metering devices. The rate is adjusted to that required to carry away any heat or fumes that

may be generated by the processing operations being carried on in this line of boxes. The inert gas inlet line is protected with an absolute type filter, and the withdrawal effluent is also passed through an absolute type filter. It is then directed into the main ventilation exhaust system, upstream from the final filters, thus ensuring that all ventilation effluents from the areas of high contamination within the glove boxes are filtered twice by absolute type filters before being released to the environment. A continuous air sampler is located in the ventilation exhaust duct (stack). It is placed sufficiently far into the duct to avoid eddy current effect of the duct walls and the flow rate is adjusted to be isokinetic with the normal exhaust velocity from the facility. The air is constantly sampled and once a week the sample is taken to the laboratory and analyzed whenever the facility is operating.

12.11 The ventilation data for each area is as follows:

1. PROCESSING AREA

   Room volume: 66,700 cu ft

   Exhaust rate: 6800 CFM

   Air changes/hour: 6

2. STILL ROOM

   Room volume: 6400 cu ft

   Exhaust rate: 1380 CFM

   Air changes/hour: 13

3. CHANGE ROOM AND TOILETS

   Room volume: 5400 cu ft

   Exhaust rate: 600 CFM

   Air changes/hour: 7

4. VESTIBULE

   Room volume: 840 cu ft

   Exhaust rate: 100 CFM

   Air changes/hour: 7

5. OFFICE TEN

   Room volume: 1260 cu ft

   Exhaust rate: 270 CFM

   Air changes/hour: 13

6. OFFICE ELEVEN

Room volume: 1260 cu ft

Exhaust rate: 270 CFM

Air changes/hour: 13

7. REST AREA TWELVE

Room volume: 1260 cu ft

Exhaust rate: 270 CFM

Air changes/hour: 13

8. OFFICE THIRTEEN

Room volume: 1260 cu ft

Exhaust rate: 270 CFM

Air changes/hour: 13

9. LABORATORY FOURTEEN

Room volume: 2270 cu ft

Exhaust rate: 300 CFM

Air changes/hour: 8

10. LABORATORY FIFTEEN

Room volume: 2270 cu ft

Exhaust rate: 2000 CFM

Air changes/hour: 53

11. OFFICE TWO TEN

Room volume: 2270 cu ft

Exhaust rate: 2000 CFM

Air changes/hour: 53

12. OFFICE TWO ELEVEN

Room volume: 2270 cu ft

Exhaust rate: 2000 CFM

Air changes/hour: 53

10.12 Other Systems--The only other auxiliary system that is of concern to nuclear safety is the Evacuation Alarm System. This system has an emergency wet battery power supply so that both the gamma detectors and the radiation alarm are operative in the event of a power failure. The modes of operation of the Evacuation Alarm System are described below:

NOTE: The exterior Nuclear Alarms have a distinctive modulating siren sound to differentiate from the interior General Emergency alarm system.

Modes of Operation

1. An event is monitored at either of two places in Building 16-A. A monitored event would cause the monitor to sound and blink. System will operate indefinitely from 115V A.C. house power or 12 hours monitoring and 20 min. alarm from battery.

Building 16-A:	Siren sounds outside
Building 16, 16-A:	Evacuation alarm inside
Building 20:	Siren sounds outside
Building 1:	Four sirens sound outside
Building 1, Civil Def. Rm:	Red light comes on indicating "bldg. 16-A alram"
Building 1, Civil Def. Rm:	Alarm silenced by key operated switch. Red light comes on indicating "Alarm Silenced"

2. An event is monitored at Bldg. 20. A monitored event would cause the monitor to sound and blink. System will operate indefinitely from 115V A.C. house power or 12 hours monitoring and 20 min. alarm from battery.

Building 16-A:	Siren sounds outside
Building 20:	Siren sounds outside
Building 20:	Evacuation alarm inside
Building 11:	Evacuation alarm inside
Building 16-A, Rm. 10:	Red light comes on indicating "Bldg. 20 alarm"
Building 16-A, Process Area:	Red light comes on indicating "Bldg. 20 alarm"
Building 1:	Four sirens sound outside
Building 1, Civil Def. Rm:	Red light comes on indicating "Bldg. 20 alarm"
Building 1, Civil Def. Rm:	Alarm silenced by key switch. Red light indicates "Alarm Silenced"

3. Test either of two monitors in Bldg. 16-A with a radiation source. Key switch in Bldg. 1, Civil Defense Room would be turned to "Alarm Silenced" position. The monitor will sound and blink.

Building 1, Civil Def. Rm:	Red light on indicating "Alarm Silenced"
Building 1, Civil Def. Rm:	Red light comes on indicating "Bldg. 16-A alarm"

4. Test monitor in Bldg. 20 with a radiation source. Key switch in Bldg. 1, Civil Def. Room would be turned to "Alarm Silenced" position. The monitor will sound and blink.

Building 1, Civil Def. Rm:	Red light on, indicating "Alarm Silenced"
Building 1, Civil Def. Rm:	Red light comes on indicating "Bldg. 20 alarm"

5. In case of a non-nuclear event at any one of Buildings 1, 1A, 2, 3,4,16,16-A and 20, the evacuation alarm would be manually tripped. Each local evacuation alarm would operate independently from house power in this mode.

Each radiation alarm is tested with a radiation source by Health and Safety and the emergency power supply is checked for charge once each month. Each unit is tagged with the date of the last inspection and the initials of the inspector. Action will be taken to repair faulty equipment immediately upon detection.

12.13 Building 20 Storage--Building 20 is a storage warehouse and inside of it, an area 16 feet by 20 feet has been fenced with chain link fence which reaches to the ceiling. The area is used for storage of material in DOT approved shipping packages only. Storage will also be in accordance with other DOT restrictions such as the maximum number per array, depending upon the types of material and concentration of U in each container. Packages are moved in and out of this area which is secured by lock in the manner specified in the accountability manual. No operations are performed in this area which could violate the safety inherent in the shipping packages; however a radiation alarm sensor is located on the wall at one side of the storage area. The operation of this alarm and its relation to the alarm system is described in paragraph 12.12.

Appendix A

Extrapolations of Critical Configurations for Increased Enrichments

The material buckling for uranium metal varies with percent U-235, according to the following formula from DP 532 page 210

$$B^2 = 0.0009323\ y - 0.00361$$

where

$$y = \text{percent U-235}$$

The material buckling Bm^2 for 93.2% enrichment for which extensive experimental criticality data are available is thus

$$B^2 = 0.000932 \times 93.2 - 0.00361$$
$$= 0.0833\ cm^{-2}$$

and for 97% enrichment

$$B^2 = 0.0009323 \times 97 - 0.00361$$
$$= 0.0868\ cm^{-2}$$

The difference in the critical thickness of an infinite slab would be as follows:

$$Bm^2 = \frac{\pi^2}{(T + 2S)^2}$$ where T is critical slab thickness

S = 4.1 cm for full water reflection

DP-532 Figure 2.2 .

then

$$T = \frac{\pi}{Bm} - 2S$$

for 93.2% $= \frac{3.1416}{0.0833} - 8.2 = 10.88 - 8.2 = 2.68$ cm

for 97% $= \frac{3.1416}{0.0868} - 8.2 = 10.66 - 8.2 = 2.46$ cm

and the critical slab thickness is reduced by just under 8% in going from 93.2% to 97% U-235.

A fully reflected infinite slab of solution 93.2% U-235 has a minimum critical thickness of 1.8 inches from TID 7028 Figure 11. Reducing this thickness by 8% to account for increased U-235 concentration of 97% results in a critical slab thickness of 1.66 inches. The slab thickness operating limit for solutions of Table 9.4 is 1.5 inches which is 90% of critical thickness if fully reflected. The slabs in question are catch pans with only incidental reflection from the top and this reduced reflection increases the margin of safety substantially. The slab thickness limitation being only about 60% of critical on the same basis as above.

The minimum critical cylinder diameter at this density from Fig. 10 of TID 7028 is 4.2 inches for 93.2% U-235 fully reflected. Reducing this diameter by 4% as above the minimum critical diameter for 97% metal in water would be 4 inches. For UO_2 with its lower density, the void space would be correspondingly smaller, leaving less space for moderator. Therefore, the critical diameter would be slightly greater than 4 inches. When moderation control is not present, dense particles are not permitted in containers larger than 3.5 inches in diameter, which is less than 87% of the minimum critical diameter.

For the case of dense particles at 97% U-235 enrichment, we have a maximum bulk density of 70%, U content of 88.2% maximum as UO_2 which has a theoretical density of 10.9 g/cc and the U-235 density = 10.9 x 0.7 x 0.882 x 0.97 = 6.52 g/cc.

At densities less than that of metal, the spread in buckling with enrichment change becomes less in the range of densities of concern to the establishment of safe geometry in this facility.(See Table IV.15, DP 532).

Infinite Cylinder

The difference in cylinder diameter for a cylinder of infinite length for 93.2 and 97% U-235 is as follows:

$$B^2 = \frac{(2.405)^2}{(R + S)^2}$$

$$R = \frac{2.405}{Bm} - S$$

and the critical radius for 93.2% U-235

$$R = \frac{2.405}{0.2886} - 4.1 = 4.23 \text{ cm}$$

diameter = 8.46 cm

for 97% enrichment, the critical radius (R_{97})

$$= \frac{2.405}{0.2946} - 4.1 = 4.06 \text{ cm}$$

diameter = 8.12 cm

The difference is 8.46-8.12 = 0.34 cm and in percent, the diameter is reduced only 4% by increasing the enrichment from 93.2% to 97% U-235.

The minimum critical diameter for water reflected solutions of 93.2% U-235 in infinite cylinders is 5.4 inches. Reduced by 4% for increased enrichment of 97% U-235 results in a minimum critical fully reflected cylinder of 5.2 inches. The uranium containing columns in the facility are 5 inches in diameter so they would be subcritical even if fully reflected. The columns are, in fact, bare and they are spaced well apart from each other and from reflecting structures, which reduces their reactivity substantially.

Appendix B
INTERACTION CALCULATIONS

I Storage of 3.5 inch outside diameter by 9 inch long canisters.

A. Configuration

1) Each individual rack will hold six canisters in a vertical column. The total length of the column is 69 inches. The maximum length of column occupied by uranium is 54 inches. Each canister is 9 inches inside length and is constructed of 3 inch schedule 5 stainless steel pipe. The wall thickness is 0.083 inches.

2) The storage area consists of two rows of six racks each spaced 18 inches on centers in the row and 24 inches on centers between rows. A railing surrounds the rack so that traffic cannot pass closer than 15 inches from a canister. A glove box is located at one end of the storage area with 15 inches from the back of the box to the nearest canister. This box may contain one canister or similar 3.5 inch diameter container. All other stations in the area are at least 5 feet from the closest unit in the storage area.

B. Single Unit Reactivity

This area is used for the storage of particles in various stages of processing and as a practical matter, almost all

of the canisters will contain unmoderated material, but since it is not desirable to restrict the storage area to unmoderated material, the reactivity determination here assumes optimum moderation. The area is not subject to flooding and the canisters have 0.083-inch thick stainless steel walls.

Reflector savings varies inversely as the U-235 density and are reported to be 2.15 cm for full density metal and 2.55 cm for bare stainless steel cylinders of UO_2F_2 solution at 0.538 g/cc (DP 532 Fig. 2.7 and Table IV.5). The density of the fissile material stored in the rack is intermediate; therefore, since neutrons that are reflected are not available for interaction for the purpose of interaction computation it is conservative to use S=2.55 cm.

The largest buckling occurs at the greatest density so the U-235 density for fully dense particles which is 6.52 g/cc is used. The six canisters in the rack are all considered to be one continuous unit 54 inches in length in this calculation.

The critical diameter of an infinite cylinder from Figure 10--TID 7028 for metal at 6.5 g/cc is 6.2 inches. Using 6.2 inches reduced by 4% to account for the 97% enrichment discussed above a buckling may be calculated.

$$B^2 = \frac{(2.405)^2}{(R + S)^2} \qquad R = 7.56 \text{ cm}$$

$$S = 2.55 \text{ cm}$$

$$B^2 = 0.0566 \text{ cm}^{-2}$$

The buckling for the rack taken as a 3.33 inch inside diameter by 54 inch long column is

$$Bg^2 = \frac{(2.405)^2}{(R + S)^2} + \frac{\pi^2}{(x + 2S)^2}$$

where

$R = \frac{3.33}{2} \times 2.54 = 4.23$ cm

x = length = 54 x 2.54 = 137.2 cm

$S = 2.55$ cm

$$Bg^2 = 0.1258 + 0.0005$$
$$= 0.1263 \text{ cm}^{-2}$$

K_{eff} may then be computed as follows:

$$K_{eff} = \frac{1 + M^2 Bm^2}{1 + M^2 Bg^2}$$

letting

$M^2 = 38$ cm which is applicable to a highly moderated system

then

$$K_{eff} = \frac{1 + 38 \times 0.0566}{1 + 38 \times 0.1263} = 0.54$$

M^2 increases with decreasing moderation and K_{eff} decreases, therefore:

$$K_{eff} \lesssim 0.54$$

The permissable interaction solid angle for a K_{eff} of 0.54 is 3.7 steradians (Figure 26--TID 7016 Rev. 1).

C. Solid Angle Interaction in the Array

By inspection of the Storage Area layout it can be seen that the distances to the glove boxes is so great at 5 feet from the nearest box to any storage unit that the most central unit of the array should be the most reactive by reason of interaction with its much closer neighbors. This interaction is computed using the formula of TID 7016 Rev.1 where $\Omega = (2d/h) \sin \theta$

No. of Neighbors Seen	Distance inches	Solid angle per Neighbor	Total Solid Angle
2	16.5	0.344	0.688
1	22.5	0.227	0.227
2	28.8	0.158	0.316
2	42.5	0.084	0.168
		Total	1.399

say, 1.4 steradians
Total permissable 1.8 steradians

Such other units as there are in the room contribute less than 0.05 steradians each and may therefore be neglected.

II Liquid Storage Interaction

A. Configuration

All uranium solutions in the process other than waste solutions are stored or treated in a series of 5 inch schedule 40 PVC columns which are located in one vertical plane. The individual tanks are separated from each other by 24 inches on center and any one column seès only its adjacent neighbors. These columns vary in length from 140 to 160 inches and they may therefore be treated as infinite.

There are no other containers or tanks containing more than 5 g/l fissile material at any time sufficiently close to subtend a solid angle greater than 0.05 steradians.

B. Single Unit Reactivity

The solution concentration in the columns is about 2 molar in uranium. As a maximum, the material buckling for a UO_2F_2 solution of 93.5% U-235 at 2.14 molar is given in Table IV.15 of DP-532 as $Bm^2 = 0.03166\ cm^{-2}$. If this buckling is increased by 4%, it will approximate the buckling of 97% enrichment and $Bm^2_{97} = 0.03293\ cm^{-2}$.

The bare column of 5.047 inch inside diameter has a 1/4 inch wall and a reflector savings S = 4.0 cm fairly represents the reflection experienced by the column.

then:

$$r = 6.41\ cm$$

$$Bg^2 = \frac{(2.405)^2}{(r + S)^2} = \frac{(2.405)^2}{(6.41 + 4.0)^2} = 0.05337\ cm^{-2}$$

$$Keff = \frac{1 + M^2Bm^2}{1 + M^2Bg^2} = \frac{1 + 38 \times 0.03295}{1 + 38 \times 0.0534}$$

$$\underline{\underline{Keff = 0.74}}$$

The permissable solid angle from Figure 26 TID 7016 Rev. 1 is 1.5 steradians.

C. Interaction Solid Angle

By the point to plane method of TID 7016 Rev. 1 $= \frac{2d}{h} \sin\theta$ where d = 5.047 inches; h = 27.0-2.5 = 24.5 inches and $\sin\theta = 1$ then

= 0.412 steradians for each column and the central column sees its two neighbors giving a total solid angle of 0.824 steradians. The maximum permissable angle is 1.5 steradians and a large margin of safety exists.

III Particle Columns

The particle forming columns contain on the average, a very low concentration of fine particles since each column produces less than a critical mass of particles each hour and by reason of the relatively high density of the particles compared to the drying liquid they quickly sink into the reduced diameter section at the bottom of the column. The solid angle between these two low activity columns is only 0.36 steradians in any event.

IV Waste Boildown System

The waste boildown system consists of two six inch schedule 40 pipes, spaced 27 inches on center,for storage of boildown feed. Also present in the same plane is the boildown evaporator, the outside of which is six inch pipe, and the boildown unit which has a six inch diameter disengaging section and smaller four inch and two inch diameter liquid sections. The boildown feed normally contains only trace quantities of fissile material. The concentrated solution is removed in five inch bottles to birdcage storage and therefore, it is highly unlikely that any of the solution in this array will exceed minimum critical concentration. The total solid angle between the boildown unit which will be the most reactive and the column it sees, is 0.5 steradians.