APPLICATION FOR AEC LICENSE TO
RECEIVE, POSSESS, USE AND TRANSFER
SPECIAL NUCLEAR MATERIAL

Pursuant to Code of Federal Regulations,
Title 10 - Atomic Energy, Part 70 -
Special Nuclear Material

TO: United States Atomic Energy Commission
Washington 25, D.C.
Attn: Division of Licensing and Regulations

Note: Item numbers and principal terminology agrees directly with pertinent item numbers of paragraph 70.22 of the subject Federal Regulations.

(1) Description of Applicant:

Name Coors Porcelain Company

Address 600 Ninth Street
Golden, Colorado

Incorporated in the State of Colorado

Principal Office is 600 Ninth Street, Golden, Colorado

Principal Officers:

President: Joseph Coors

(b)(6)

FILED NOV 18 1963 U.S. Atomic Energy Commission DIVISION OF LICENSING & REGULATION

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Executive Vice President:
William K. Coors
c/o Adolph Coors Company
Golden, Colorado
(b)(6)

Secretary: Ray V. Frost
(b)(6)

Assistant Secretary:
Morris B. Hecox
(b)(6)

Treasurer: Adolph Coors, Jr.
c/o Adolph Coors Company
Golden, Colorado
(b)(6)

There is no control or ownership exercised over Coors Porcelain Company by any alien, foreign corporation, or foreign government.

(2) The general activity, for which special nuclear material will be requested is the development and manufacture of reactor fuel in various forms and configurations.

This application is made for licensing of this facility on a capability basis rather than for a particular activity.

For the purpose of clarification, one type of activity, to which the contemplated license would apply, may be the manufacture of fuel rods for use in the Special Power Excursion Reactor Test (SPERT) facility, which is operated for the Commission by Phillips Petroleum Company. A proposal for this work is presently being submitted by Coors Porcelain Company to Phillips Petroleum Company.

The place at which all contemplated activity will be performed, to the extent that the handling of special nuclear material is directly involved (See Note 2), is the Energy Products Division, located in the Fuel Element Building (SS Station CPC), at the firm's principal address.

The exact physical location of the Fuel Element Building (at 7th and Ford Streets in Golden, Colorado) is clarified by the following engineering drawings which are included in License Application Supplement No. 1, which is submitted with this application:

Coors Porcelain Company Drawing No. 45-0022-23
Adolph Coors Company Drawing No. 44-0039-7
Adolph Coors Company Drawing No. 44-0039-34

Note 2: It is possible that unfueled mechanical parts of fuel element assemblies (such as vessels, brackets, etc.) may be partly or completely fabricated in locations other than the one described in this application, when such manufacturing steps can be completed prior to contact with special nuclear material.

(3) The period of time for which the license is requested begins immediately and extends until December 31, 1969.

(4) The name, amount, and specifications of the special nuclear material which Coors Porcelain Company proposes to use are as follows:

Name -Contained U-235

Amount - The total quantity of contained U-235 on the property at any one time will not exceed 1,000 kilograms (contained in UO_2 and U_3O_8).

Specifications - The material may be mixed with a 95% ThO_2-5% UO_2 combination or may be processed as UO_2 alone.

The chemical form will be UO_2 and/or U_3O_8 in one or more of the following physical forms during processing:

Powder
Slurry
Mixed with organic wax or binder
Pressed into pellets
Calcined
Sintered in hydrogen atmosphere
Grinding sludge

(5i) The estimated date on which Coors Porcelain Company desires to receive the first shipment of special nuclear material is (on or before) January 2, 1964.

(3) The period of time for which the license is requested begins immediately and extends until December 31, 1969.

(4) The name, amount, and specifications of the special nuclear material which Coors Porcelain Company proposes to use and/or produce are as follows:

Name -Contained U-235

Amount - The total quantity of contained U-235 on the property at any one time will not exceed 1,000 kilograms (contained in UO_2 and U_3O_8).

Specifications - The material may be mixed with a 95% ThO_2-5% UO_2 combination or may be processed as UO_2 alone.

The chemical form will be UO_2 and/or U_3O_8 in one or more of the following physical forms:

Powder
Slurry
Mixed with organic wax or binder
Pressed into pellets
Calcined
Sintered in hydrogen atmosphere
Grinding sludge

(5i) The estimated date on which Coors Porcelain Company desires to receive the first shipment of special nuclear material is (on or before) January 2, 1964.

Subsequent receipts of special nuclear material, for the uses presently anticipated, are estimated to occur within the calendar years of 1964, 1965, 1966, 1967 and 1968.

(5ii) A schedule, by years, showing estimated consumption and operating losses of special nuclear material can not be projected beyond 1964, except to suggest that the example used for 1964 is generally typical of a part of the work anticipated for this facility.

Again referring the the Special Power Excursion Reactor Test (SPERT) facility, it is estimated that 200 to 500 kilograms of contained U-235 will be consumed[1] during the calendar year 1964, and that operating losses are not estimated not to exceed 5 kilograms of contained U-235 during the same period.

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[1] The terms "consumed" or "consumption" include the reduction in value of material due to blending of different assays of special nuclear material, or other alternation of the isotopic ratio, and the disposition of material in such manner that it cannot be economically recovered for further use.

(5iii) An estimated schedule, by years, for the transfer of special nuclear material to the U. S. Atomic Energy Commission and to other licensees is subject to the same considerations stated in the preceding Paragraph (5ii).

In the event that the SPERT fuel rod contract is awarded to Coors, approximately 400,000 grams of contained U-235 will be transferred to Phillips Petroleum Company by approximately November 15, 1964, based upon an estimated final delivery date for SPERT fuel rods.

Supporting data for the above estimates consists of the following specifications, which were taken from the Phillips Petroleum Company Request For Quotation for Subcontract No. C-255:

9000 rods at 36-inch active fuel length.

grams of UO_2 / inch = 25 grams

enrichment (U-235 = 2 to 5% U-235)

(6) The technical qualifications of the principal Energy Products Division staff members are as follows:

at Coors Porcelain Company as Design Engineer, designed the Automatic Trimming Machine and Internal Drying Oven for the Coors Automated Aluminum Can Plant, supervised construction of the present (second) aluminum can production line. Currently directing all Energy Product Division technical supporting services.

In addition to the Division staff listed, technically qualified ceramic, stainless steel, aluminum and metallizing experts are available from other Coors Divisions.

(7) Descriptions of the equipment and facilities which will be used by Coors Porcelain Company to protect health and minimum danger to life or property, are furnished in License Application Supplements Nos. 3, 4 and 8, which are submitted with this application.

Supplement Number 3 is the Health Physics Guide.

Supplement Number 4, entitled "Health and Safety Procedures" also includes some references to equipment.

Monitoring equipment is discussed in License Application Supplement No. 8.

(8) The proposed procedures to protect health and minimize danger to life or property, including procedures to avoid accidental conditions of

B. L. MORNIN, DIVISION MANAGER

Education: B. S. and M. S. degrees in Ceramic Engineering from Missouri School of Mines.

Special Training: Criticality Theory and Practice Course at the University of California, Berkeley.

Experience: Research Engineer and Production Superintendent at Electric Auto Lite Company Spark Plug Division; Research Engineer for Aluminum Company of America on manufacture of ceramic grade alumina materials; introduced the dry press forming method to the ceramic seal industry; 9 years at Coors Porcelain Company - 4 years as Plant Production Superintendent, formed Energy Products Division in 1960.

KEN G. WASSON, DIVISION SUPERINTENDENT

Education: B S. degree in Mechanical Engineering from the University of Wisconsin.

Special Training: Special Weapons, U.S.A.F., Pilot, U.S.A.F., various management and technical courses, University of Colorado and U.S.A.F.

Experience: Design Engineer, Research Test Engineer, Caterpillar Tractor Company; Special Weapons Accountable Officer, U.S.A.F.; Pilot U.S.A.F.; Test Facilities Engineer, Sundstrand Turbo (Missile Application Work); Design Engineer, Production Supervisor, Production

Superintendent, Coors Porcelain Company. Developed and designed most of the production equipment used to manufacture the fuel elements used in the Tory II-C reactor core.

C. E. NORDQUIST, SUPERVISOR
PRODUCT DEVELOPMENT, PROCESS CONTROL

Education: B. S. degree in Ceramic Engineering, University of Washington, M. B. A. degree in Management, University of Denver.
Experience: 3 years in the U.S. Air Force working on development of ceramic materials and processes for radar nose cones (Radomes) to be used in advanced vehicles (directed Air Force efforts in this area for 1-1/2 years); six (6) months in New Products Division of Coors Porcelain Company, 2 years in production trouble shooting of alumina materials; with Energy Products Division since November 1962.

R. D. SMITH, HEALTH PHYSICIST

Education: B.A. degree (1950) in Chemistry and Biology from the University of Colorado.
Special Training: Additional academic training in nuclear radiation physics, industrial toxicology, industrial ventilation and electronic instrumentation. A two month training period with the Los Alamos Scientific Laboratories in radiation monitoring procedures, conducted by the USAEC. Industrial Ventilation Conference, Michigan States University.

Experience: 8-3/4 years in Health Physics as a Radiation Engineer with the Dow Chemical Company, Rocky Flats Plant. Areas of responsibility included supervision of all Health Physics activities in facilities processing uranium, plutonium and americium, as well as conducting environmental site surveys. Team Captain of Radiological Assistance Plan Emergency Monitoring Team while at Rocky Flats. 3 years with Coors Porcelain Company in charge of health physics and nuclear safety functions, assisting in design of the Coors Energy Products facility and directing health physics functions during Lawrence Radiation Laboratory Subcontract 165 for the production of beryllia-urania fuel elements.

GEORGE BIDINGER, NUCLEAR PHYSICIST

Education: B. S. (1957) and M. S. (1958) in Physics from John Carroll University, C eveland, Ohio.

Special Training: 20 week course in Dynamics of Supervision.

Experience: 3-1/2 years at the Dow Chemical Company's Rocky Flats Division in the Nuclear Safety Group. Functions were nuclear safety evaluations of processes and equipment in the storage, transportation, processing and fabrication of highly enriched uranium and plutonium, and critical mass surveys using transport (DSN) and diffusion theory computation codes. 1 year at the Coors Porcelain Company's Energy Products Division in charge of nuclear safety for the production of beryllia-urania fuel elements for the Tory II-C reactor.

C. R. McMULLEN, CHIEF CHEMIST

Education: B. Ed degree in Chemistry from Eastern Illinois State College.

Special Training: 5 years training on the Linde Air Products Company Chem and Met Program.

Experience: 3 years as Analytical Chemist on the Manhattan Project for Linde Air Products Company in Tonawanda, New York. One year as Analytical Chemist in the Schenectady Works Laboratory, and three years as Research Chemist at the General Electric Hanford Works. 2 years for Phillips Petroleum Company at the National Reactor Testing Station as Research Chemist in the Chemical Development Laboratory. 4 years with Sundstrand Corporation, three years as Chief Chemist, and one year as Supervisor of the Chemical and Metallurgical Laboratories at the Denver Plant; 3-1/2 years at Energy Products Division.

FRANCIS S. DeROSE
ASSISTANT CHIEF CHEMIST

Education: B. S. degree (1937) in Chemistry from Regis College, Denver, Colorado. Graduate work in Chemistry and Chemical Engineering at the University of Colorado and Ohio State University.

Experience: Quality control studies at Remington Arms Ballistic Laboratory; 3 years on Manhattan Project at University of Chicago, Analytic Chemistry of Isotopes; 9 years as Professional Chemist for Dow Chemical Company, including 5 years in charge of Rocky Flats

Plant Spectrographic Laboratory; 3 years at Coors Porcelain Company, Energy Products Division as Director of Spectro-Chemistry.

T. J. LaROCCO, CHIEF INSPECTOR
ENERGY PRODUCTS DIVISION

Education: Petroleum Engineering degree from Colorado School of Mines, 1947.

Special Training: U. S. Navy Electronics School, and U. S. Navy School for Encoding and Decoding Equipment.

Experience: 10 years as Design Engineer for Stearns-Roger Manufacturing Company, designing metal reduction, chemical and cement plants. Supervised design and development of all manual inspection equipment used for Tory II-C reactor core. Member of Coors Porcelain Company staff for 3 years as director of all inspection operations related to production of urania-beryllia fuel elements.

GEORGE F. KNOWLES, PROJECT ENGINEER

Special Training: Machine Tool Operation and Toolmaking (1938)

Experience: 4 years as Manufacturing Process Engineer for Alpha Engineering Company (Mt. Prospect, Illinois) as a contractor to Ford Motor Company for design of production lines; 3 years at Sundstrand Corporation, Machine Tool Division (Belvedere, Illinois) as transfer line specialist (Ford, Chrysler, Peugot projects); 3 years at Coors Porcelain Company as Design Engineer, designed the Automatic Trimming Machine and Internal Drying Oven for the Coors Automated Aluminum Can Plant, supervised construction of the present (second) aluminum can production line. Currently directing all Energy Product Division technical supporting services.

In addition to the Division staff listed, technically qualified ceramic, stainless steel, aluminum and metallizing experts are available from other Coors Divisions.

(7) Descriptions of the equipment and facilities which will be used by Coors Porcelain Company to protect health and minimum danger to life or property, are furnished in License Application Supplements Nos. 3, 4 and 8, which are submitted with this application.

Supplement Number 3 is the Health Physics Guide.

Supplement Number 4, entitled "Health and Safety Procedures" also includes some references to equipment.

Monitoring equipment is discussed in License Application Supplement No. 8.

(8) The proposed procedures to protect health and minimize danger to life or property, including procedures to avoid accidental conditions of

criticality and procedures for personnel monitoring and waste disposal, are furnished in License Application Aupplements Nos. 3 and 4.

Supplement Numbers 5 and 6, "Nuclear Safety Considerations" and "A Typical Ceramic Production Process" will provide helpful background information for clarifying many of the health protection measures.

The probability of special nuclear materials being involved in a fire in this facility are very remote. The building structure is generally of a fire-proof nature, and process materials at all times, are in a non-combustible state and no unusual fire hazards are believed to exist.

One area which does present unusual potential hazards is the high temperature, hydrogen atmosphere furnaces used for sintering. These furnaces have been in operation on almost a continual basis for nearly two years and have not given any indication of malfunction which would create a fire or explosion. Entrance and exit of materials is accomplished by automated inter-locked controls which purge the entrance and exit chambers with nitrogen gas thereby preventing the mixture of outside air with the hydrogen atmosphere inside of the furnace. Frequent surveys are made for leakage of hydrogen gas or any malfunction.

Nearly all of the air exhausted from the building is exhausted through our process equipment ventilation system. This air is passed through absolute type filters before being released to the outside atmosphere. Should an incident occur within the building causing the release of radioactive material into the atmosphere within the building, this air would not represent a serious potential release to the outside environment.

It is highly improbable that any incident occurring within the confines of the Fuel Element Building would cause a significant release of radioactive atmosphere to the outside environment.

Regarding financial responsibility, the Coors Porcelain Company was incorporated in 1911 in Colorado and has continued in business up to the present time.

In the three fields of ceramic manufacturing;

Chemical and Scientific Ware,

Grinding (ceramic) Media, and

Industrial Alumina parts,

Coors is considered as the leader in this country.

Since this is a family owned corporation, and financial statements are not published, the rating given by Dun and Bradstreet is (1). This indicates net worth of $125,000 to $1,000,000. Annual sales approximate 10 million dollars. We refer you to the First National Banks of both Golden and Denver, Colorado for verifying this Company's integrity and ability to adequately finance the license requested.

Coors recently completed an A.E.C. contract which totaled approximately $5,200,000. This also, we believe, will evidence our ability to handle and finance the license applied for.

Note: The following item numbers agree directly with pertinent item numbers of paragraph 70.24 of the subject Federal Regulations.

(a1) The radiation monitoring system is discussed in License Application Supplement No. 8.

(a2) Emergency Procedures are discussed in License Application Supplement No. 3 on page 20, 45 through 51, and in Supplement No. 4 on pages 45 through 51.

(b) Plans for compliance by Coors Porcelain Company with the requirements of this section (Reference: paragraph 70.24, "Additional Requirements") are considered to be in effect as of the date of this application.

A complete statement of these plans is deemed to be included as part of License Application Supplements Nos. 3, 4 and 8.

CERTIFICATE

(This item must be completed by applicant)

The applicant, and any official executing this certificate on behalf of the applicant named in Item 1, certify that this application is prepared in conformity with Title 10, Code of Federal Regulations, Part 40, and that all information contained herein, including any supplements attached hereto, is true and correct to the best of our knowledge and belief.

Dated November 14, 1963

COORS PORCELAIN COMPANY

BY: Joseph Coors

PRESIDENT

(Title)

FORM AEC-2
(2-61)
Previous editions are obsolete.

FORM APPROVED
BUREAU OF BUDGET NO. 38-R002.8

UNITED STATES ATOMIC ENERGY COMMISSION

APPLICATION FOR SOURCE MATERIAL LICENSE

Pursuant to the regulations in Title 10, Code of Federal Regulations, Chapter 1, Part 40, application is hereby made for a license to receive, possess, use, transfer, deliver or import into the United States, source material for the activity or activities described.

1. (*Check one*)
- [x] (*a*) New license
- [ ] (*b*) Amendment to License No. ______
- [ ] (*c*) Renewal of License No. ______
- [ ] (*d*) Previous License No. ______

2. NAME OF APPLICANT

Coors Porcelain Company

3. PRINCIPAL BUSINESS ADDRESS

600 Ninth Street
Golden, Colorado

4. STATE THE ADDRESS(ES) AT WHICH SOURCE MATERIAL WILL BE POSSESSED OR USED

Fuel Element Building (SS Station CPC) 7th and Ford St., Golden, Colorado

5. BUSINESS OR OCCUPATION

Industrial Ceramics

6. (*a*) IF APPLICANT IS AN INDIVIDUAL, STATE CITIZENSHIP

Colorado Corporation

(*b*) AGE

7. DESCRIBE PURPOSE FOR WHICH SOURCE MATERIAL WILL BE USED

See: Application for AEC License to Receive, Possess, Use and Transfer Special Nuclear Material, Paragraph (2)

8. STATE THE TYPE OR TYPES, CHEMICAL FORM OR FORMS, AND QUANTITIES OF SOURCE MATERIAL YOU PROPOSE TO RECEIVE, POSSESS, USE, OR TRANSFER UNDER THE LICENSE

(*a*) TYPE	(*b*) CHEMICAL FORM	(*c*) PHYSICAL FORM (*Including % U or Th.*)	(*d*) MAXIMUM AMOUNT AT ANY ONE TIME (*in pounds*)
NORMAL URANIUM	UNH	Crystals 55.3% U	660 lbs.
URANIUM DEPLETED IN THE U-235 ISOTOPE	U_3O_8	Powder 84.7% U	10,000 lbs.
	UO_2	Powder 88.1% U	15,000 lbs.
THORIUM	ThO_2	Powder 87.8% Th	13,000 lbs.

(*e*) MAXIMUM TOTAL QUANTITY OF SOURCE MATERIAL YOU WILL HAVE ON HAND AT ANY TIME (*in pounds*)

30,000 lbs.

9. DESCRIBE THE CHEMICAL, PHYSICAL, METALLURGICAL, OR NUCLEAR PROCESS OR PROCESSES IN WHICH THE SOURCE MATERIAL WILL BE USED, INDICATING THE MAXIMUM AMOUNT OF SOURCE MATERIAL INVOLVED IN EACH PROCESS AT ANY ONE TIME, AND PROVIDING A THOROUGH EVALUATION OF THE POTENTIAL HAZARDS ASSOCIATED WITH EACH STEP OF THOSE OPERATIONS.

See License Application Supplements Nos. 3 through 6.

10. DESCRIBE THE MINIMUM TECHNICAL QUALIFICATIONS INCLUDING TRAINING AND EXPERIENCE THAT WILL BE REQUIRED OF APPLICANT'S SUPERVISORY PERSONNEL INCLUDING PERSON RESPONSIBLE FOR RADIATION SAFETY PROGRAM (OR OF APPLICANT IF APPLICANT IS AN INDIVIDUAL).

Refer to Application for AEC License To Receive, Possess, Use and Transfer Special Nuclear Material, Paragraph (6). Also see License Application Supplement No. 13, Statement No. 1.

11. DESCRIBE THE EQUIPMENT AND FACILITIES WHICH WILL BE USED TO PROTECT HEALTH AND MINIMIZE DANGER TO LIFE OR PROPERTY AND RELATE THE USE OF THE EQUIPMENT AND FACILITIES TO THE OPERATIONS LISTED IN ITEM 9; INCLUDE: (*a*) RADIATION DETECTION AND RELATED INSTRUMENTS (including film badges, dosimeters, counters, air-monitoring and other survey equipment as appropriate. The description of radiation detection instruments should include the type of radiation detected and the range(s) of each instrument.)

Refer to Application for AEC License to Receive, Possess, Use and Transfer Special Nuclear Material, Paragraph (7).

(*b*) METHOD, FREQUENCY, AND STANDARDS USED IN CALIBRATING INSTRUMENTS LISTED IN (*a*) ABOVE (for film badges, specify method of calibrating and processing, or name supplier.)

Refer to License Application Supplement No. 4

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11(c). VENTILATION EQUIPMENT WHICH WILL BE USED IN OPERATIONS WHICH PRODUCE DUST, FUMES, MISTS, GASES, ETC.

Refer To License Application Supplement No. 3 and 4.

12. DESCRIBE PROPOSED PROCEDURES TO PROTECT HEALTH AND MINIMIZE DANGER TO LIFE AND PROPERTY AND RELATE THESE PROCEDURES TO THE OPERATIONS LISTED IN ITEM 9; INCLUDE:
(a) PROCEDURES FOR USE OF NUCLEAR MATERIALS AND SAFETY FEATURES AND PROCEDURES TO AVOID NONNUCLEAR ACCIDENTS, SUCH AS FIRE, EXPLOSION, ETC., IN SOURCE MATERIAL STORAGE AND PROCESSING AREAS.

Refer to License Application Supplements 3, 4, and 5.

(b) EMERGENCY PROCEDURES IN THE EVENT OF ACCIDENTS WHICH MIGHT INVOLVE SOURCE MATERIAL.

Emergency Procedures are discussed in License Application Supplement No. 3 on page 20, 45 through 51, and in Supplement No. 4 on pages 45 through 51.

(c) DETAILED DESCRIPTION OF RADIATION SURVEY PROGRAM AND PROCEDURES.

Refer to License Application Supplement No. 3 and 4.

13. WASTE PRODUCTS: *If none will be generated, state "None" opposite (a), below. If waste products will be generated, check here* ☒ *and explain on a supplemental sheet:*
(a) Quantity and type of radioactive waste that will be generated.
(b) Detailed procedures for waste disposal.

Ref: License Application Supplement No. 13 Statements 2 and 3.

14. IF PRODUCTS FOR DISTRIBUTION TO THE GENERAL PUBLIC UNDER AN EXEMPTION CONTAINED IN 10 CFR 40 ARE TO BE MANUFACTURED, USE A SUPPLEMENTAL SHEET TO FURNISH A DETAILED DESCRIPTION OF THE PRODUCT, INCLUDING:
(a) PERCENT SOURCE MATERIAL IN THE PRODUCT AND ITS LOCATION IN THE PRODUCT.
(b) PHYSICAL DESCRIPTION OF THE PRODUCT INCLUDING CHARACTERISTICS, IF ANY, THAT WILL PREVENT INHALATION OR INGESTION OF SOURCE MATERIAL THAT MIGHT BE SEPARATED FROM THE PRODUCT.
(c) BETA AND BETA PLUS GAMMA RADIATION LEVELS (*Specify instrument used, date of calibration and calibration technique used*) AT THE SURFACE OF THE PRODUCT AND AT 12 INCHES.
(d) METHOD OF ASSURING THAT SOURCE MATERIAL CANNOT BE DISASSOCIATED FROM THE MANUFACTURED PRODUCT.

CERTIFICATE

(This item must be completed by applicant)

15. *The applicant, and any official executing this certificate on behalf of the applicant named in Item 1, certify that this application is prepared in conformity with Title 10, Code of Federal Regulations, Part 40, and that all information contained herein, including any supplements attached hereto, is true and correct to the best of our knowledge and belief.*

COORS PORCELAIN COMPANY
(Applicant named in Item 1)

Dated November 14, 1963

BY: Joseph Coors

PRESIDENT
(Title of certifying official authorized to act on behalf of the applicant)

WARNING: 18 U.S.C. Section 1001; Act of June 25, 1948; 62 Stat. 749; makes it a criminal offense to make a willfully false statement or representation to any department or agency of the United States as to any matter within its jurisdiction.

DOCKET NO. 70-814

LICENSE APPLICATION SUPPLEMENT NO. 5
November 14, 1963

Nuclear Safety Considerations

This material is furnished for reference use as a supplement to Coors Porcelain Company applications for AEC licenses.

RECEIVED
NOV 18 1963
U.S. Atomic Energy Commission
DIVISION OF
LICENSING & REGULATION

NUCLEAR SAFETY CONSIDERATIONS

I. Introduction

In the many operations associated with the development and production of fuel components and assemblies, the hazards of a criticality accident are some of the more important considerations. This guide summarizes the limitations and operating techniques which will be in effect for the prevention of this type of accident. It is being prepared primarily for the benefit of the nuclear safety control group; however, it may also assist management in the safe operation of development and production facilities.

The criteria are generally consistent with the nuclear safety standards of the Nuclear Safety Guide.(1) Data from other critical mass experiments may be used with caution.

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(1) Nuclear Safety Guide, Rev. 1, TID-7016, 1961 revised by the Subcommittee 8 of the American Standards Association Sectional Committee W6 and Project 8 of the American Nuclear Society Standards Committee.

II. Philosophy of Nuclear Safety Control

A. General

1. Nuclear Safety Control is basically concerned with the protection of Energy Products Division personnel, plant facilities, and the surrounding communities from the hazards of radiation and contamination associated with a criticality accident.

2. Maximum effort is directed to the prevention of criticality accidents. However, such accidents are possible and consequently the protection includes plans for handling an accident of this type.

B. Responsibilities

Nuclear safety is a line responsibility; that is, each supervisor is responsible for the practices of his subordinate supervisions and reporting personnal as well as the mechanical aspects of each area.

1. Process supervision has the following responsibilities:
   a. To be adequately acquainted with all aspects of operations which involve nuclear safety and to assure that all operations are carried out according to approved procedures.

b. To request the advice of Nuclear Safety personnel concerning any unusual circumstance or condition which might affect nuclear safety.

c. To request the advice of Nuclear Safety personnel concerning contemplated changes in equipment or operating procedures.

2. Nuclear Safety personnel have the following responsibilities:

a. From a nuclear safety point of view, to advise process supervision on the design, installation, alteration and operation of equipment and facilities for processing, storing and transporting U-235.

b. To assist in the investigation of non-routine incidents such as infractions and near-misses of nuclear safety rules, and to suggest possible methods of eliminating such incident in the future.

c. To assist supervision in the design and layout of instrumentation for the detection of radiation from a critical mass accident, should one occur.

. To provide a program of nuclear safety lectures and demonstrations to those groups requesting such service and to new personnel.

C. Criteria

1. Criteria for nuclear safety control are based on the results of criticality experimentation. In the absence of such experimentation, conservative reductions of existing criteria are made.

2. Calculations in conjunction with experimental data may be used to bridge the gap between experimental points. Criteria are not based on calculations alone, but may be used as a back-up verification.

3. Nuclear safe geometry is preferred for nuclear safety control over the other methods of control.

4. Administrative control on a nuclear safe mass basis will be used for safe operation of non-geometrically safe equipment in most cases where nuclear safe geometry is impractical.

5. Fixed neutron poisons (Pyrex glass, Cadmium foil, etc.) may be used as a primary means of control in some circumstances.

6. An operation is considered safe if it requires the simultaneous failure of two independent safeguards for the establishment of an unsafe uranium configuration. Sabotage is not considered a factor in establishing criteria for a nuclearly safe operation.

D. Emergency Planning

1. Radiation detection instruments with automatic alarms for the detection of any criticality are installed at intervals within the department where fissionable material is handled. For further description of this system see License Application Supplement No. 4, page 45.

2. A comprehensive plan of action is maintained in order to cope with a criticality accident. This plan is explained in detail in License Application Supplement No. 4, page 45.

III. Nuclear Safety Criteria

For U-235 Moderated with Hydrogeneous Materials and for Unmoderated U-235

A. Basic Criteria

The criteria in Table I are recommended as being nuclearly safe for U-235 when moderated and reflected with light water.[1]

TABLE I

	Recommended Nuclearly Safe for Solutions
Mass-Kg: $(H/U \leq 2)$	2.0
Mass-Kg: $(H/U > 2)$	0.35
Diameter of Infinite Cylinder - inches:	5.0
Thickness of Infinite Slab - inches:	1.5
Solution Volume - liters	4.8

B. <u>Handling Rules</u>

The following handling rules are applicable for:

1. *Non-moderated uranium* (H/U-$235 \leq 2$):

   a. Individual batches of U-235 are $\leq$ 2000 g.

   b. The minimum separation between batches, which is maintained by physical spacers, is 1 foot edge to edge.

   c. Positive controls are in effect to prevent water and other moderating materials from being mixed with the fuel.

2. Moderated uranium (H/U-$235 > 2$) or where No. 1. c. above does not apply:

   a. Individual batch sizes are $\leq$ 350 g.

   b. The limitation in No. 1. b. above is in effect.

C. <u>In-Process Storage Rules</u>

The following precautions for storage are nuclearly safe for non-moderated and moderated U-235 batches in planar arrays. Although water flooding is not credible, the separation between batches is safe for flooded or unflooded conditions.

1. Individual containers are covered and essentially water tight.

2. Individual containers are in fixed positions while in storage.

3. Individual containers are physically separated by a minimum of 12 inches edge to edge.

4. Individual containers are on minimum center to center spacing of 16 inches.

GHB:br

LICENSE APPLICATION SUPPLEMENT NO. 6

A Typical Ceramic Production Process

This material is furnished for reference use as a supplement to Coors Porcelain Company applications for AEC licenses.

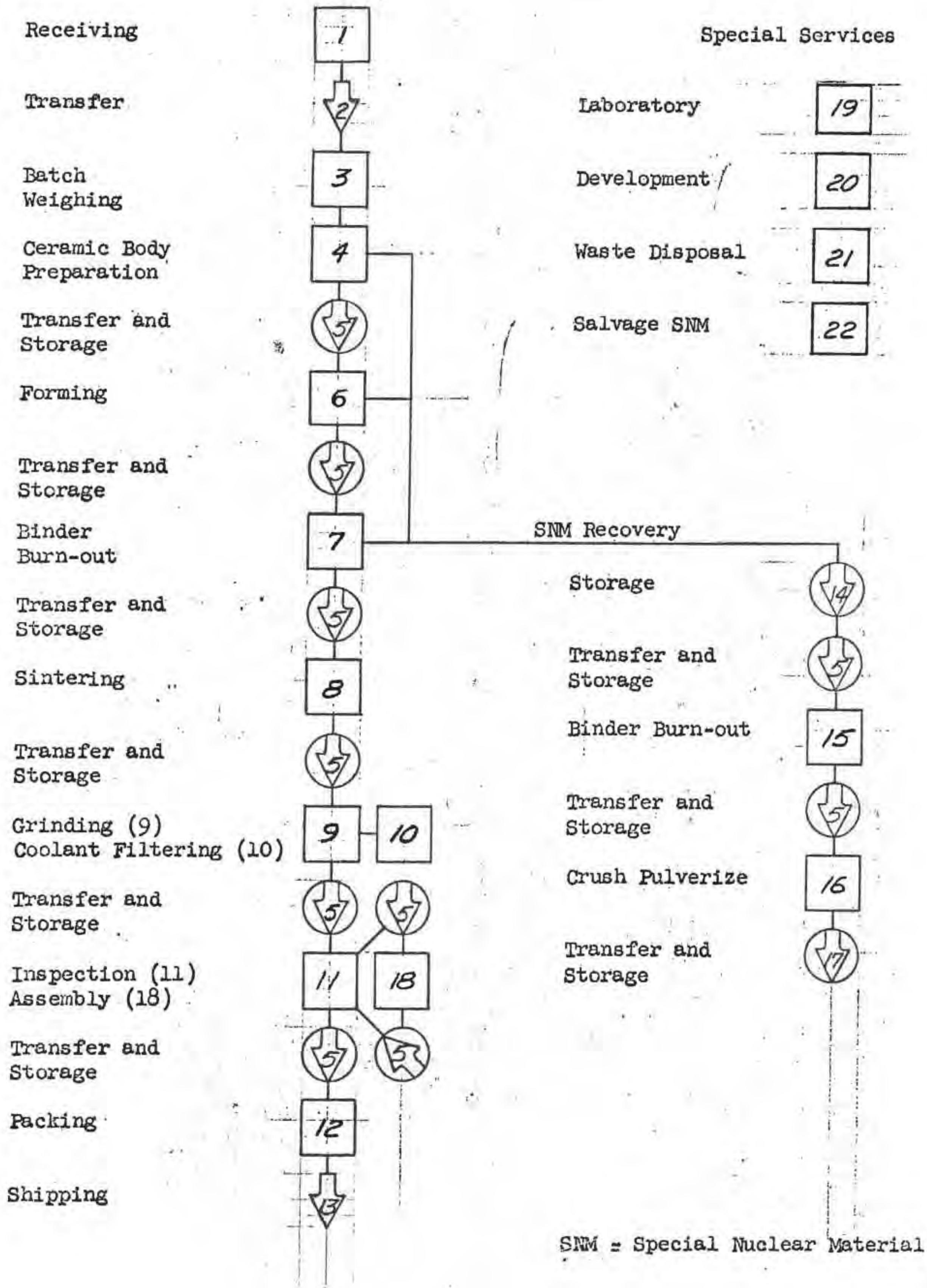

FIGURE 1 - TYPICAL PROCESS FLOW DIAGRAM

Introduction

This facility has been operating under a more detailed Nuclear Safety Guide for the production of urania-beryllia fuel elements under subcontract to the University of California. The more detailed plan, as approved by AEC-SAN and UCLRL, is documented as CPC-68, classified Secret - Restricted Data. Copies of this document are available at AEC-SAN, UCLRL, and Coors Porcelain Company. Because of the security restrictions, it is not intended to license the classified equipment, which is used exclusively for our present subcontract work for LRL.

Layout of Production Equipment

Coors Porcelain Company Engineering Drawing No. AE-4-581-1, Revision 5 (1) shows the first floor layout of production equipment. This level is above ground and flooding is not credible.

At the west end of the building is a section known as the Depressed Area. This area is below grade and could be flooded. Waste disposal equipment (See Paragraph 21 below) is located in this area.

Drawing No. AE-4-528-2, Revision 1 shows the second floor layout of production equipment and the laboratory equipment.

Typical Process

The process flow diagram in Figure 1 depicts a typical ceramic production process to be performed in the Energy Products Division facility. This

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(1) Prints of applicable engineering drawings are included in License Application Supplement No. 1.

diagram is used for illustrative purposes and is not intended to represent the only capability of this facility.

The numbers on the flow diagram refer to the paragraph numbers in the following text in which the respective operations are described.

1. Receiving

The SNM is received from the commercial carrier at the dock area by a member of the accountability staff. The material in the shipping birdcages is either moved into vault storage or stored in nuclear safe protected arrays in the dock area. The size and shape of the array depends upon the chemical and physical forms of the SNM and on the number and size of the shipping birdcages.

The material is then taken to the sampling station where the material is weighed and a sample is withdrawn. The SNM is then returned to the vault to await transfer to the Production Department.

The sample is taken to the laboratory to check for uranium content, impurities, moderation, assay, etc.

2. Transfer

A the need for more SNM arises in the Production Department, a member of the accountability staff transfers the material (still in the shipping birdcage) to the Production Department.

Nuclear safety is assured by using the birdcage as a transport medium and by limiting the amount of material being transferred to that quantity which is in one birdcage.

3. Batch Weighing

The SNM is weighed into batch quantities. The maximum size batch which is independent of enrichment or moderation is 350 grams of U-235. For low enrichments and low well-controlled moderation, the batch size can be increased to a maximum of 2000 grams of U-235. This increase would be done in accord with criteria and curves in Figure 1, Figure 2, Figure 3, Figure 4, Figure 20, and Figure 21 in the Nuclear Safety Guide (1a).

Nuclear safety is controlled by:

a. Placing the batch in a restricted use container.

b. Identifying the SNM with a batch card.

c. Moving the batch to an approved storage location.

d. Limit of contained U-235.

e. Geometry of process equipment.

In addition to the batch processing, process control samples are withdrawn from various stages of the operation which are limited in any collection of samples to the weights allowed for one batch. Personal conveyance inside the facility of up to 20% of a batch limit is allowed without further nuclear safety restriction.

(1a) Nuclear Safety Guide, Revision 1, TID-7016, 1961 revised by the Subcommittee 8 of the American Standards Association Sectional Committee W6 and Project 8 of the American Nuclear Society Standards Committee.

4. Ceramic Body Preparation (2)

The SNM batch is mixed with binder, water, lubricant, additives, etc. in blenders, mixers, or roller mills. This mixing equipment, by design, is good for batch operation only. The batch size as measured above in Paragraph 3 is determined by the amount of moderation added at this process step.

Administrative control is used to insure that only one batch is at a mixing station at one time.

5. Transfer and Storage (3)

Hand carrying of batches is generally avoided. The batch in a proper container is wheeled in a transport cart to a storage rack designed for that container. The cart is designed to keep the batch 6 inches from the edge of the cart. The batch card remains with the batch to identify the quantity and type of material in the container.

Nuclear safety is enhanced by using a "safe" transport cart and properly identifying the contents of the batch with the batch card. For batch size, see Paragraph 3 above.

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(2) For additional information on this step, see Photograph Number 22-5 in License Application Supplement No. 2.

(3) For photographs which help to illustrate these operations, see Photographs Numbers 22-3, 22-4, 22-6, 22-8, 22-13 and 22-14 in License Application Supplement No. 2.

5. Transfer and Storage continued

Storage racks are constructed to maintain minimum 12-inch edge to edge and 16-inch center to center spacings between batches. Physical barriers fill the spaces between the "pigeonholes." Guard rails are erected a minimum of 6-inches in front of pigeonholes.

Six inch spacing is used on the carts and on the storage rack guard rail so that 12-inch separation is maintained between batches on carts and in storage racks.

Piping is being considered as a mode of transport for SNM in a fluidized medium-air or liquid. All pipe line and pipe connections will be nuclear safe by geometry. This geometry includes pipe connections, interconnections, and SNM fluidized medium separators.

6. Forming (Extrusion)

Forming can be done by the extrusion method. Two large extruders are used for production and two additional small extruders are used for pre-extrusion testing.

The pre-extruders have bore sizes of not greater than 2-inches in diameter. These extruder stations are limited to one batch each.

The two production extruders have 4.8-inch diameter by 2-foot long bores. These are nuclear safe for more than one batch. The material is split into batch size or smaller quantities and the extruded parts are placed on saggers for Binder-Burn-Out (See Paragraph 7 below).

Forming (Dry Press)

A mechanical press can be used to form parts from "dry" ceramic SNM. The ceramic material is described as being "dry" because of its appearance; however, the binder will usually be a hydrogen compound.

A glove box for loading and unloading the press tools will also be employed. (This box does not yet exist.) Two nuclear safety controls will be used for this semi-automatic process - batch control and safe geometry.

The parts as pressed will be loaded on saggers for firing as mentioned in Paragraphs 7 and 8 below. The height of the material on the saggers will be limited so that only a safe array can be formed in the furnaces.

7. Binder Burn-Out

A Hevi-Duty electric furnace is used to remove all binder products. The ceramic parts are placed on saggers after forming. The saggers enter a heated zone with a cross-sectional opening $6\frac{1}{2}$-inches by $13\frac{1}{2}$-inches. However, the current saggers can hold material in a 1 3/4-inch by 11-inch cross-sectional area. The saggers are butted together to form essentially a 1 3/4 by 11-inch infinite array in the furnace. Different saggers could be used for different products. The height of the array for different enrichments or assays could also change.

Binder Burn-Out continued

Geometry is used to control nuclear safety. The furnace is an open air furnace which permits visual inspection for administrative control.

8. Sintering

The ceramic parts are sintered in electrically heated hydrogen atmosphere furnaces. The chamber is approximately 6-inches high by $7\frac{1}{4}$-inches wide. Saggers are butted together to form essentially an infinitely long array. The height of the ceramic material is controlled by administrative control to form a safe planar array. The height of the array is a function of the assay and moderation of the SNM.

9. Grinding (4)

Each grinding station consists of a grinder, a work table, and a coolant cart.

Actual fuel grinding operations are conducted inside of dry boxes. The grinding coolant returns to the coolant cart by gravity.

Each work table is a two-level table with a one foot thick physical barrier between levels. A well is located in the centers of each of the two levels so that a batch cannot come within one foot of the edge of the table or each other.

(4) Also see Photograph Number 22-7 in License Application Supplement No. 2.

The coolant carts are approximately 13 gallon capacity, three inch thick annulli. The 14-inch diameter inner cylinder is cadmium lined and water filled.

The well in the top level of the work table and the grinder is considered a station for one batch. A second batch may be in the well on the lower level of the work table.

10. Coolant Filtering (5)

The coolant carts are nuclear safe cylindrical annulli. The outside of the cylinder is 12-inches from the edge of the cart; the 14-inch diameter inside cylinder is cadmium lined and water filled. The annulus is 3-inches thick and has a 13 gallon capacity.

The coolant and grinding sludge is dropped through the floor in the grinding department to a 21-inch diameter by 3-inch high filter pan. The filtrate is collected in pyrex pipe filled holding tanks until the filtrate is analyzed. The filtrate is dumped to the sewer if Health Physics requirements are met. If the filtrate cannot be dumped to the sewer, it is refiltered until requirements are met.

The sludge is calcined in batch quantities or less as described in Paragraph 15 below. The sludge is disposed in the manner described in Paragraph 22 below.

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(5) Also see Photograph Number 22-9 in License Application Supplement No. 2.

11. Inspection

The inspection department has cleaning, dimensional check, zyglo check, X-ray check, and gamma scanning equipment. Each individual piece of inspection equipment is designated as a station. Only one batch is permitted at a station. Where more than one station occupies the same table or work bench, a one foot thick barrier separates the work stations.

Administrative control is used to limit the station to one batch. See Paragraph 3 above for batch size.

12. Packing

The Special Nuclear Material products are packaged in shipping containers which are water tight and nuclearly safe when water reflected. Bird cages are used to maintain a minimum of twelve inches, edge to edge, between shipping containers. Design of shipping containers will be approved by Bureau of Explosives prior to usage.

13. Transfer

The birdcages are stored in the dock area for transfer to a commercial carrier, generally an exclusive-use vehicle.

14. Special Nuclear Material Recovery Storage

In each process area, some Special Nuclear Material from each batch, which can be rêcovered and reprocessed, is scrapped. This material is composited into a recovery batch under carefully controlled conditions. Chemical analysis, total weight, and/or volume are used to determine the composite batch limit.

15. Special Nuclear Material Recovery Binder Burn-Out

Batches of Special Nuclear Material for recovery are placed in 9-inch pans for removal of all binder products. The batch material is limited to an approximate depth of 2-inches, which is necessary for good binder burn-out. These pans are placed edge to edge and run through the furnace described in Paragraph 7 above. The pans in effect form a 2-inch by 9-inch by infinite array of Special Nuclear Material.

A small batch type furnace is also available for binder burn-out.

16. Special Nuclear Material Recovery Crush and Pulverize

This operation is a four step operation in a recovery "silo" dry box. The Special Nuclear Material to be recovered is gravity fed through a jaw crusher, down through a pulverizer, down through a dry ferro filter, and into a chute for bag-out. This is a batch type operation with only one batch allowed in the dry box which houses all of this equipment. Each batch is weighed in and out of the silo to prevent any unknown buildup of material in the silo.

17. Special Nuclear Material Recovery Transfer and Storage

The batch is ready for operations described in Paragraph 3 above. See Paragraph 5 for Transfer and Storage procedures.

18. Assembly

Assembly of ceramic parts will be with batch quantities or less Special Nuclear Material unless the material is placed in a geometrically safe shape. The assembly will be handled in a birdcage except for certain inspection operations which prohibit the use of the birdcage.

19. Laboratory

The analytical equipment available includes two Arc-Spark Emission Spectrograph (3 meter and 1-1/2 meter), a 50KV X-ray Fluorescent Emission Spectrograph, and a wet chemistry laboratory.

This equipment is sufficnet for purity analysis, assay analysis, and some isotope analysis (± 1% between 5% and 95% U-235 enrichment).

The nuclear safety limits for the laboratory are one batch equivalent in the Arc-Spark area, one batch equivalent in the X-ray area, and one batch equivalent in the wet chemistry area (three batch equivalents total). A batch equivalent is defined as the number of batch samples which, if combined, would be equal to one batch. See Paragraph 3 for batch size.

20. Development

One end of the wet chemistry laboratory has been designated as the process development area.

One batch equivalent of material is allowed in this area. See Paragraph 3 for batch size.

21. Waste Disposal[6]

Contaminated liquids (20 ppm uranium) are collected in two 1000-gallon tanks located in the Depressed Area. When a tank is full of liquid, it is flocculated, filtered, sampled, and dumped to the sewer if permitted by Health Physics regulations. If dumping is not permitted, it is again flocculated, filtered, etc., until it can be dumped to the sewer. The sludge is collected for burial in 30-gallon drums. Contaminated solids (-20 ppm uranium) are collected in 55-gallon drums for burial.

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(6) Also see Photograph Number 1-12 in License Application Supplement No. 2.

22. Salvage Special Nuclear Material

All Special Nuclear Material which can be salvaged is stored in 5-inch cylinders in birdcages. This material includes laboratory samples, grinding sludge and sintered scrap. A maximum of 1 Kg of U-235 will be collected in any cylinder.

The 5-inch cylinders in birdcages (55-gallon drums) will be stored in one of the following arrays prior to shipment for recovery:

a. A group of 50 in any pattern. Each group will be separated by 12-feet from other Special Nuclear Material.

b. A group 2 wide, one high and infinitely long. Each group will be separated 12-feet from other Special Nuclear Material.

LICENSE APPLICATION SUPPLEMENT NO. 7
November 14, 1963

Statement of Combined Operation

Approximately 110 Kg of uranium enriched in U-235 is accounted for at CPC under subcontract to LRL. The nuclear safety plan for this material is documented in the CPC Nuclear Safety Guide, CPC-68. It has been approved by AEC-SAN and the LRL Nuclear Safety Committee. This material would at no time be mixed with any licensed material which would come into the CPC facility. This non-mixing includes all production, inspection, and storage areas.

CPC - Coors Porcelain Company
LRL - Lawrence Radiation Laboratory

LICENSE APPLICATION SUPPLEMENT NO. 8
November 14, 1963

Radiation Monitoring System

A five channel remote area radiation monitor monitoring system is currently in operation at the Energy Products Dvision facility. This system was installed and approved as part of Coors Porcelain Company's subcontract work for the Lawrence Radiation Laboratory.

At this time, the system is being re-evaluated in the light of the requirements set forth in paragraph 70.24 of CFR 70. In consideration of the many new pieces of equipment and re-arrangement of the production area, which may be needed in conjunction with activities carried out under this license, it is necessary to re-consider the radiation sensitivity of each detector at its present location.

Rather than delay the submission of license applications while these calculations are made, it is intended to submit the applications without the detailed description of our Gamma Monitoring System while the above calculations are completed.

LICENSE APPLICATION SUPPLEMENT NO. 13
November 14, 1963

Miscellaneous Supporting Statements

1. In liew of a description of minimum technical qualifications, the actual backgrounds of responsible supervisory personnel are stated.

2. Discussion of the quantity of radioactive waste that will be generated is limited by the intent to license the capability of the Energy Products Division facility, as explained in Application for AEC License to Receive, Possess, Use and Transfer Special Nuclear Material, Paragraph (2).

3. See License Application Supplement No. 6, Figure 1 and Paragraph No. 21.

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LICENSE APPLICATION SUPPLEMENT NO. 4
November 14, 1963

DOCKETED
USAEC
Division of
Licensing & Regulation
NOV 1 8 1963
Docket Officer

Health and Safety Procedures

This material is furnished for reference use as a supplement to Coors Porcelain Company applications for AEC licenses.

7666

4.2.1.3 HEALTH AND SAFETY PROCESS

4.2.1.3.1 AIR SAMPLING

General Air Samples

The Giraffe portable air samplers (Gast Pump Model 0521) are located with respect to the operation, air currents, and personnel and can be moved to any location. Air samples are collected on 5.4 cm diameter Millipore paper backed by a 5.4 cm diameter Whatman #41 filter paper by placing the plastic sample head, containing the Millipore and Whatman #41 filter paper, into the connector on the neck of the pump. An air sample is collected for 24 hours at the breathing level of the workmen. Before turning the pump on, an elapsed time and clock time reading is taken and recorded on the air sample record. The air sample record card is attached to the pump.

Data Gathering

At the completion of the sampling period, the pump is turned off and the sample number, location, date, time on, time off and personnel present are recorded on the air sample record card.

Total time is calculated by subtracting the elapsed time readings giving the time to the nearest tenth of an hour. Multiply by sixty to obtain time in minutes. This is converted to minutes and entered on the total time line.

To obtain the volume in M^3, multiply the flow rate (normally 2 cfm) by 0.0283 and the result by the total time in minutes.

Duct Samples

There are eight permanent duct air sampling stations located throughout the building. Main exhaust duct sample is located behind the Mechanical Technician's shop in the M & E room; pump is between fan motor and duct-work. Fuel tower exhaust duct is located at the top of the stairs on the second level of the Fuel Tower. Laboratory Hood numbers 1, 2, 3 and 4 are located above the laboratory area in the first fan complex. Laboratory glove box numbers 5 and 6 are located above the laboratory area in second fan complex.

Gast rotary air pumps (Model 0521) are used and samples collected on 5.4 diameter Millipore paper backed by a 5.4 cm diameter Whatman #41 filter paper.

The two filter papers are placed in the double air sample head so that the Millipore paper is facing the duct. The air sample is collected for seven days and is usually changed every Friday morning about 10:00 a.m.

Before turning the Gast pump on, an elapsed time and clock time reading is taken and recorded on the air sample record card which is retained by the Technician taking the sample. At the completion of the sampling period, the sample number, location, remarks, date, time on, and time off are recorded on the sample record card. Calculate the total time by subtracting the elapsed time meter readings. This gives the times in tenths of minutes. Multiply by sixty to obtain the time in minutes. Enter this figure on the total time line. The volume, in M^3 is obtained by multiplying the flow rate (normally 2 cfm) by 0.0283 and the result by the total time in minutes.

High Volume Samples - In-Plant

The Staplex Hi-volume air sampler (Model TFA41) used for sampling large volumes of air employs a turbine type blower and permits the use of 4-inch diameter filter papers. The filter paper must be placed evenly over the 4-inch opening and secured with the aluminum filter holder. The removable cross-grid is then inserted behind the filter paper and the sampler turned on. The sampler may be located wherever a sample is required. The sample number, location, date, time on, time off, and personnel present are recorded on the sample record card.

The flow rate is measured by means of an indirect orifice meter located on the back of the Staplex Hi-volume sampler which indicates the pressure drop across an orifice in the housing and is calibrated against a standard orifice on the intake. Total volume, in M^3 is found by multiplying the average flow rate, in cfm by 0.0283 and the result by total time in minutes.

Permanent Off-Site Sampling Stations

The sampling equipment for the three off-site stations consists of Gelman Nuclear Air Samplers #26001, with 5.4 cm filter holders, elapsed time meter and flow rate meter. The original Gelman Nuclear Air Sampler pumps have been replaced by Gast Rotary Air Pumps model 0521.

The three permanent off-site sampling stations are located from 100 meters to 1600 meters from the plant. Number one sampler is located on the north east corner of the Porcelain Plant roof. Number two sampler is located on the west outside wall of the Brewery Garage. Number three sampler is located on the south east corner of the Sewage Plant roof. Continuous off-site air samples from the permanent stations are collected weekly on a 5.4 cm diameter Millipore paper backed by a 5.4 cm diameter Whatman #41.

Prior to departing for the sites, the following materials must be assembled: Three air sample record cards, three air sample heads with filter papers inserted in proper order, three glassine envelopes, a dark color marking pen, 1 quart 10 SAE motor oil, (non-detergent) and the site survey books including combination to off-site locks.

At each site, the following operations must be performed: Change the air sample heads, check motor oil level in oil reservoir. To add oil, run pump while filling and leave cap off the oil reservoir for a few minutes. Check Norgren trap located behind Gast pump in housing unit. If oil has collected in the trap, drain and keep a written record in the site-survey book. Lock housing unit with combination lock. Record the sample number, location, date, time on, time off, elapsed time meter readings, flow rate meter readings weather, and approximate temperature on the air sample card and in the site-survey log book. Calculate volume of air sampled by subtracting the flow rate meter readings at start and finish of sampling period. The value in cubic feet is converted to cubic meters by multiplying by 0.0283.

High Volume Samples - Off-Site

The following equipment and supplies are required for collecting high volume off-site samples:

Hi-volume air sampler in wooden case.
Gasoline powered electric generator.
Gas can and gasoline.
Whatman #41 filter paper.
Glassine envelopes.

The following steps are carried out at each site:

Place one piece of filter paper in sample head.
Start generator and sampler.

Adjust generator power level to give a flow of about 20 cfm.
Run sampler sixty minutes.
Record weather conditions.
Record start/stop time and sampler flow volume in log book.
Place completed sample in glassine envelope for submission to Counting Technician.

4.2.1.3.2 RADIATION DETECTION

Alpha Monitoring

The Nuclear Chicago P-2112 portable alpha particle counter is used with an attached air proportional alpha probe. As Alpha particles have a very short traveling distance in air, it is necessary to hold the sensitive area of the probe as close as possible (within 1/8 inch) to the surface being monitored.

The process stations where high alpha contamination is most likely to occur should be monitored at least once per shift. These areas include:

P-10 Primary bottle weigh station
P-20 Primary bottle supply station
P-40 Fuel weigh out
All other stations where fuel is processed

At least twice per shift, the counter should be checked for calibration by removing the calibration standard from the clips on the end of the case and holding it on the center of the face of the probe. After allowing sufficient time for the needle to stabilize, the needle should read the count written on the back of the standard. If not, the instrument is calibrated by turning the "calibrate" screw <u>very slowly</u> until the meter reads correctly.

When necessary, the Probe window may be replaced in the following manner:

Stretch a piece of aluminized mylar smoothly over a felt faced block of wood about 3" x 10". Tape the mylar to the block, clean the brass window frame and put a light coat of rubber cement on one side. Lay the cemented side of the frame on the stretched mylar very carefully. Allow time for the cement to dry.

Take a sharp knife and cut, very carefully, around the outside of the frame. The window is ready for use and the probe may be re-assembled.

The Nuclear Chicago P-2112 counter is used primarily to indicate that a hazardous contamination condition exists and vigorous clean-up is required. The meter reading is usually relayed verbally to the operator concerned.

The actual count observed at a station is not recorded unless unusually high. However, the record of having conducted a survey is reported on the Nuclear Safety Monitoring Report form for that shift.

Beta-Gamma Surveys

Nuclear Measurements Corp. GS-3 is used to monitor:

Sludge buildup in the SS waste tanks.

Sludge buildup in the tank truck.

U-235 indication in pure BeO storage places.

The Victoreen AGB-50B-SR is intended primarily as a re-entry survey instrument in the event of a nuclear excursion. Neither the NMC GS-3 or the Victoreen AGB-50B-SR are used for specific data gathering but as indicators of a need for more accurate investigation. It is expected that the results from these surveys be negative, therefore, they are reported only on the Nuclear Safety Monitoring Report form as having been conducted.

4.2.1.3.3 AREA SURVEYING

Contamination Surveys

A daily survey should be made in and around all stations processing fueled material.. An alpha survey with a portable alpha survey instrument and/or a smear survey should be made in addition to the routine housekeeping smear survey. All dry box gloves, the work bench tops, machine controls, floors, and other areas of suspected contamination should be monitored.

Routine housekeeping smear surveys should be counted for alpha activity before being submitted to the laboratory for beryllium analysis.

Contamination should be reported to the area supervisor immediately and a written record of the survey made turned into the H.P. office. A contaminated incident (accident) report should be filled out when necessary. The contaminated area should be checked after decontamination to assure that proper clean-up was completed.

Nuclear Safety Surveying

At least once per shift, the following nuclear safety surveys should be made:

Check all production stations to see that no limits are being exceeded.

Check the uranium inventory in the laboratory and development area to assure that the amount of fuel present is less than the area limits.

Check all storage racks for proper material storage.

Inspect all coolant carts at the beginning of the shift to assure that there is less than the allowable amount of sludge in them. Initial the coolant log sheet only if you are sure that the limit will not be exceeded during your shift.

Check the ultrasonic water, 1000 gallon waste tanks, zyglo solutions, degreaser and tank truck with the Beta-Gamma counter. Report any readings above background.

Observe that personnel do not violate nuclear safety rules as posted by sign limits or as stated in CPC-68. Stop violations before they start. If need be, fill out a nuclear safety infraction report.

Industrial Safety Surveying

Monitor all areas to assure proper use of safety equipment and the wearing of proper safety clothing such as glasses, face shields, etc.

Make certain that all walk ways and exits are kept clear in the event that they must be used for an emergency evacuation.

Make weekly surveys of the H_2 furnaces for H_2 leaks.

Make sure that there are sufficient respirators in all areas.

Record Keeping and Reporting

Contaminated incident reports and nuclear safety infraction reports are to be completed and sent to supervision.

Routine smears, water, and soil samples are to be sent to the Counting Technician with a Chemistry Laboratory Request for Analysis.

4.2.1.3.4 RADIOMETRIC ANALYSIS

Alpha Air Samples

A SAC-2 Eberline instrument with a PC6-1 pulse counter is used for this analysis.

At the beginning of each day a geometry and background determination must be made. The pre-determined source is placed in the counting chamber, and the timer is set for ten minutes. To obtain the percentage of geometry (on the SAC-2, it should be between 39.5% and 40.5%) the total count is divided by the time, and the resultant then divided by the known number of disintegrations of the source. To raise the geometry, the bottom of the chamber is raised by rotating it counterclockwise. To lower the geometry, the bottom is rotated clockwise.

To determine the background, a clean filter paper is placed in the chamber, and the timer set for ten minutes. The total count is divided by the time counted to get the background. If the background exceeds 0.9 cpm, the chamber is washed with water or alcohol until the count is down to an acceptable level.

The air sample to be counted must be handled very carefully with forceps to avoid smudging. Care should be taken that the paper is below the level of the chamber so there is no chance of it catching on the edge when the drawer is closed. After the sample has been counted, it is placed in a glassine envelope and attached to the Air Sample Record sheet.

Routine air samples are held for a 24 hour period before counting, and are counted once for a time of ten minutes. Off-site samples are held for a 24 hour period and are counted for 20 minutes. After a minimum 4 hour interval, preferably close to 24 hours, the samples are counted a second time for 20 minutes.

To calculate the routine samples which have been counted only once, the resultant cpm is entered as cpm (LL) on the Air Sample Record form. This figure is then divided by the geometry (0.40) to yield disintegrations per minute. This figure is then divided by the volume in cubic meters as obtained from the record form. The resulting value is entered on the form as dpm/M^3, the dpm/M^3 are divided by 220, the MPC, and the resulting figure is multiplied by 100 to obtain percent MPC.

To calculate samples which have been counted two times, the Koval formula for radioactive decay correction is applied. This formula is stated as:

$$\text{cpm(LL)} = \frac{C_2 - C_1K}{1 - K}$$

where cpm(LL) are the counts per minute due to long lived isotopes,
C_2 is the value in cpm for the second count
C_1 is the value in cpm for the first count
K is a decay constant dependent on the time interval between counts and is determined from the following graph:

After the value for cpm(LL) has been obtained, it is entered on the Air Sample Record form. This figure is then divided by the geometry of the counter (0.40), to yield disintegrations per minute which is then divided by the volume in cubic meters. This value is entered in the form as dpm/M3.

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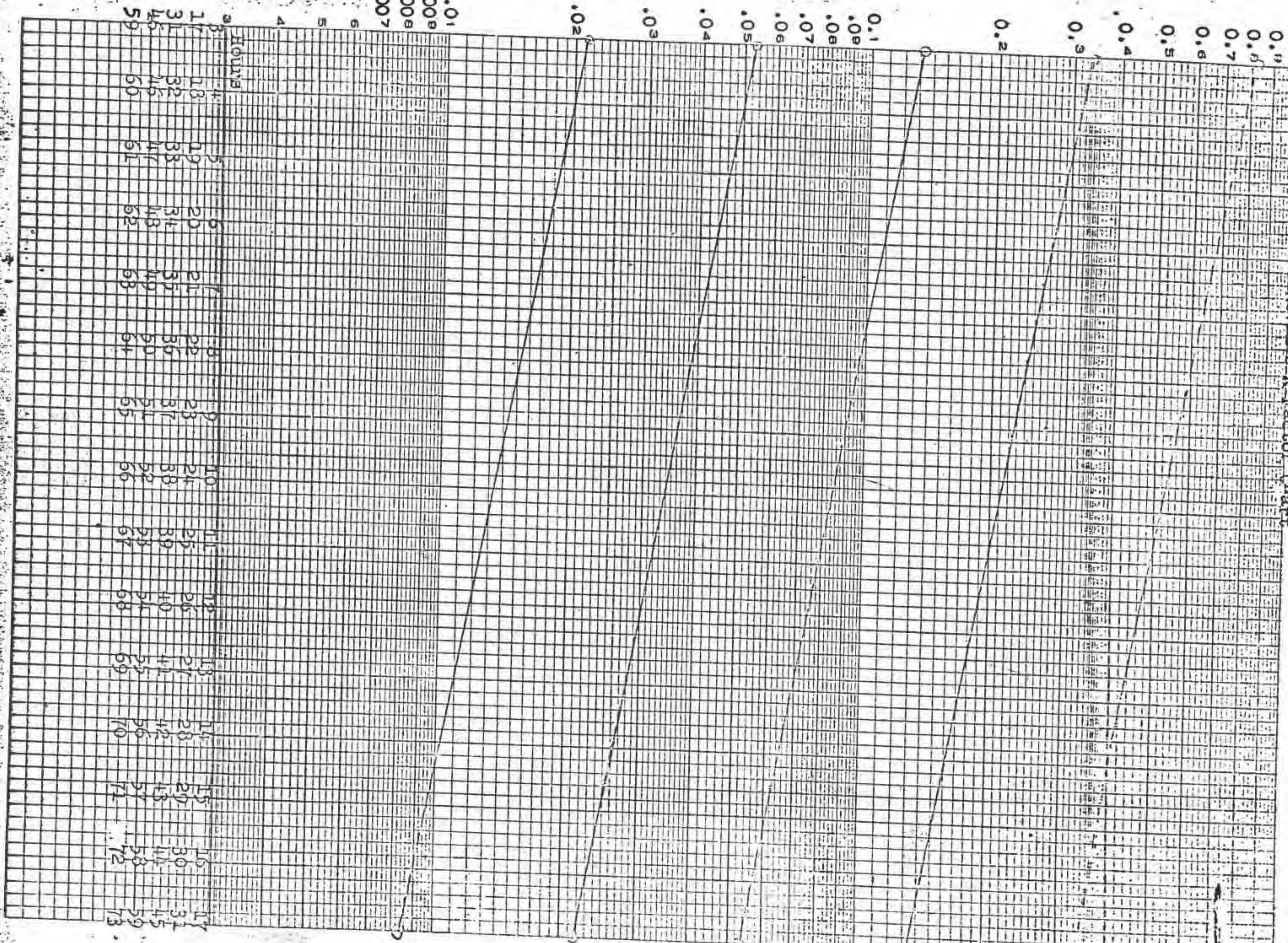


To obtain an answer in terms of percentage of Maximum Permissible Concentration for inplant samples counted in this manner, divide the dpm/M^3 value by 220, the MPC, and multiply the result by 100 to obtain %MPC.

To obtain an answer in terms of percentage of allowable concentrations for ducts and offsite locations, the dpm/M^3 value is divided by 2.2×10^{-12} which yields the concentration in microcuries per cubic centimeter. This value is entered in the form in parentheses on the same line as the dpm/M^3. The uc/cc value is then divided by the maximum allowable concentration of 4×10^{-12} uc/cc, the result multiplied by 100 and the resultant value entered as %MPC.

After alpha determination, the sample is sent to the laboratory for beryllium determination and the carbon copy of the Air Sample Record is retained in the laboratory. The original record is returned to Health Physics and is kept on file.

All in-plant, duct, and off-site samples are reported on a weekly basis to Fuel Element Supervision, CPC Engineering, and CPC Research.

Alpha Smears

The same counting equipment and methods for determining geometry and background are used as described in the preceding section.

Whatman #41, 2.8 cm paper is used for smears and must be handled very carefully to avoid rubbing off any contamination that may be present. A small paper booklet rather than a glassine envelope is used to hold the smears.

The smear paper is placed very carefully in the center of the chamber, making sure the contaminated side is up and it is below the level of the chamber so that the drawer will close without catching the smear paper. All smears are counted for alpha determination immediately for at least 30 seconds to obtain CPM, which is multiplied by 2.5 to obtain DPM.

A complete record is kept on all smears. A Request for Analysis form is completed describing the origin of the smear and the type of analysis desired. If alpha only is requested, the person submitting the smear is notified immediately of any contamination present.

AIR SAMPLE RECORD

SAMPLE NO.	DATE
LOCATION	TIME ON
	TIME OFF
REMARKS	TOTAL TIME
	FLOW RATE ______ cfm
	VOLUME ______ M^3

ALPHA DETERMINATION

	FIRST COUNT	SECOND COUNT
DATE		
TIME		
GEOMETRY		
TOTAL COUNT		
COUNTING TIME		
GROSS CPM		
BACKGROUND		
NET CPM		
TECHNICIAN		

BERYLLIUM DETERMINATION

DATE

LAB IDENTIFICATION

MICROGRAMS Be

TECHNICIAN

CONCENTRATIONS

ALPHA	BERYLLIUM
CPM (LL)	MICROGRAMS/M^3
CPM/M^3	
%MPC	%MPC

If a beryllium analysis is requested, the smear is returned to its booklet, attached to the Request for Analysis form and submitted to the Laboratory. When the Laboratory results are received, a report is then made to the person submitting the smear.

Alpha Water

The same counting equipment and methods for determining geometry and background are used as described in the preceding section.

When a water sample is submitted for analysis, a 1 ml sample is taken using a pipette, placed in a 1" planchet, and set under the heat lamp to evaporate. The planchet is placed in the counting chamber, with a clean filter paper beneath it to eliminate any chamber contamination, and counted for ten minutes. The total count is divided by the time, the background is subtracted and the result is Net CPM. This is then divided by the geometry (40%) and the result is DPM/ml. DPM/ml is divided by 2.2×10^{-6} to obtain the uc/ml.

All water samples, laundry rinse water and waste disposal water are entered in the waste water log book. The alpha results are reported on a Request for Analysis form and returned to the person submitting the sample.

Gamma Water

The counting system used for gamma analysis is a Gamma Spectrometer, consisting of a RIDL-49-55 Scaler, a Victoreen-695 Single Channel Differential Analyzer, and a Linear Pulse Amplifier-LE-1023.

At the beginning of each day a background determination must be made. A standardized vial of de-ionized water is placed in the counting chamber, and counted for ten minutes. This total count is divided by the time to obtain the background. A standard solution of known value is then placed in the chamber and counted for ten minutes to determine the accuracy of the machine. All liquids, filtrates, zyglo solutions, ultrasonic, coolant solutions are submitted for gamma analysis in a standard 60 ml polyethylene mixing vial.

Samples are normally counted for ten minutes and the total count divided by the time and the background subtracted. The Net CPM are compared to a graph made up from known standards and the proper PPM figure determined. A second procedure, to be followed when a more accurate determination is desired, or when counting anything other than a liquid, is described on the attached form.

All samples, with the exception of waste water, are entered in the Gamma Log Book. The person submitting the sample is notified of the results as soon as they are available.

4.2.1.3.5 SITE SURVEY

Water Samples

One liter water samples will be collected on a weekly basis and submitted to the laboratory for a beryllium and uranium analysis. Sample locations are shown on the accompanying map. Each week the sample is taken at a point nearest the air sample location for the week. The bottle is dated and marked with the sample location. Beryllium analysis is requested in terms of ug/liter.

Soil Samples

Seventy five cc soil samples are collected weekly at the location of the water samples and submitted to the laboratory in 100 cc plastic cold cream jars for beryllium and uranium analysis. Beryllium analysis is requested in terms of ug/gram. The jars are dated and marked with the sample location. Sample locations are indicated on the accompanying map.

Air Samples

Samples are collected from the following three permanent stations weekly:

Northeast corner of Porcelain Plant roof.

West outside wall of the Brewery Garage.

Southeast corner of the Sewage Plant roof.

Standards and unknown must be same size, shape, and concentration.
Obtain net cpm (2,4) for standards above (1) and below (3) net cpm for sample (8).
Obtain difference between concentrations (5) and net cpm (6) for standards.
Divide net cpm difference (6) by concentration difference (5) to determine ratio (7).
Subtract low standard cpm (4) from sample cpm (8).
Divide answer (9) by ratio (7) to give result (10).
Add low standard concentration (3) to result (10) to obtain Sample Concentration (11).

	conc.	units	net cpm	
High Standard (1)				(2)
Low standard (3)				(4)
(1) - (3) = (5).				(6) = (2) - (4)

$\frac{(6)}{(5)}$ = (7)

Sample net cpm(8)

-(4)

(9) / (7) = (10)

+ (3)

(11) Sample Conc.

High volume air samples are collected weekly at three of the predetermined locations off-site. The locations are shown on the accompanying map. The locations are numbered from 1 - 23 and are described by their UTM grid locations. Samples are collected at locations in numerical sequence, requiring approximately eight weeks to complete the sequence of sampling locations.

Each Hi-Vol sample is collected for a period of one nour at a flow rate of approximately 20 cfm. The sample is submitted for alpha analysis and beryllium determination.

Grid System

The Universal Transverse Mercator grid system has been selected as the most clear and precise method of locating soil, water and air sample points. This system, based on 1000 meter grids is shown in the following maps.

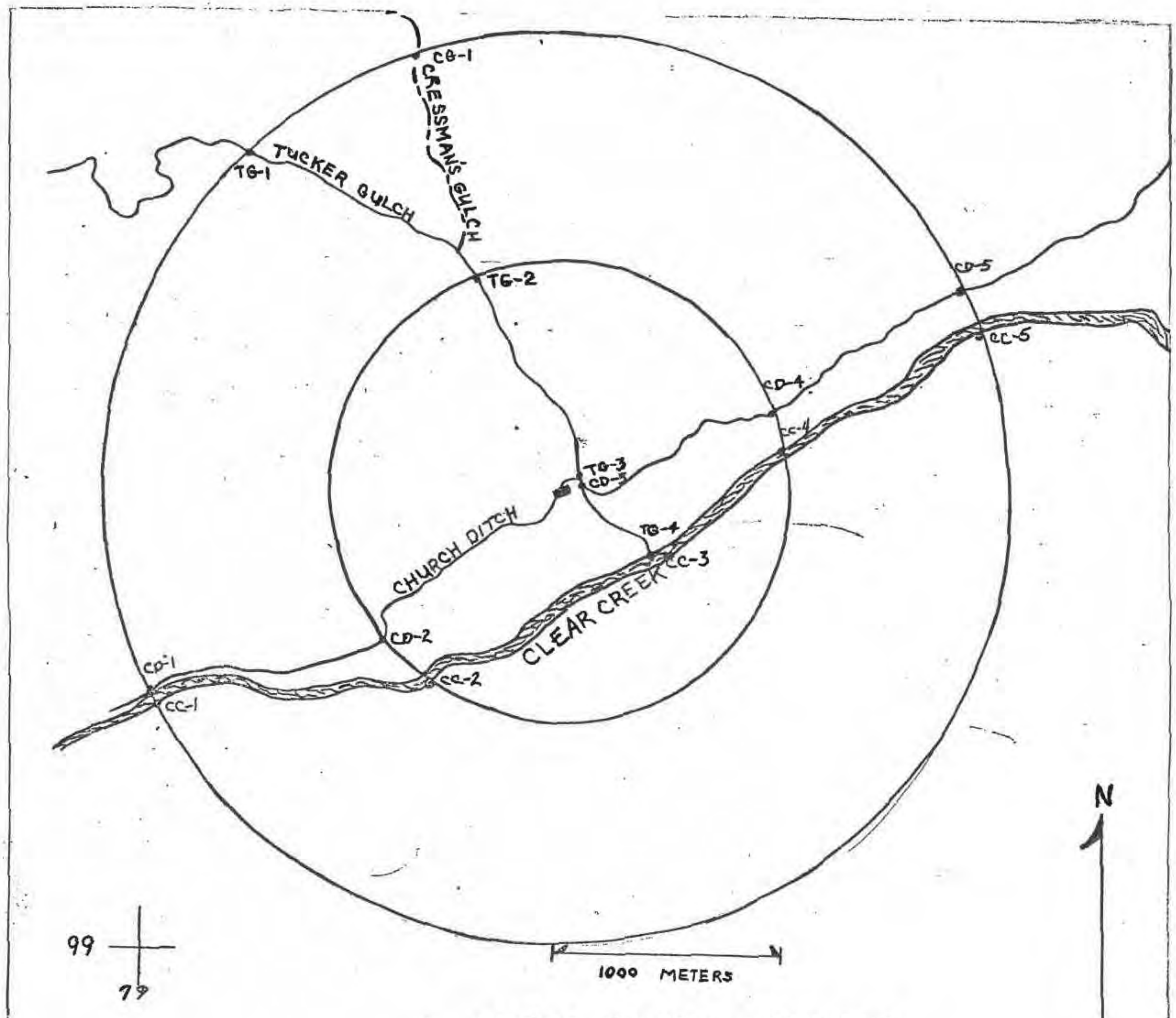
A reference coordinate of 99-79 was chosen and the grids continue from there north and east in 1000 meter increments. Sample points for the most part were located radially at 100, 500, 1000 and 2000 meters from the plant.

Each sample point was given a number and assigned geographical coordinates. These coordinates are based on the UTM projection.

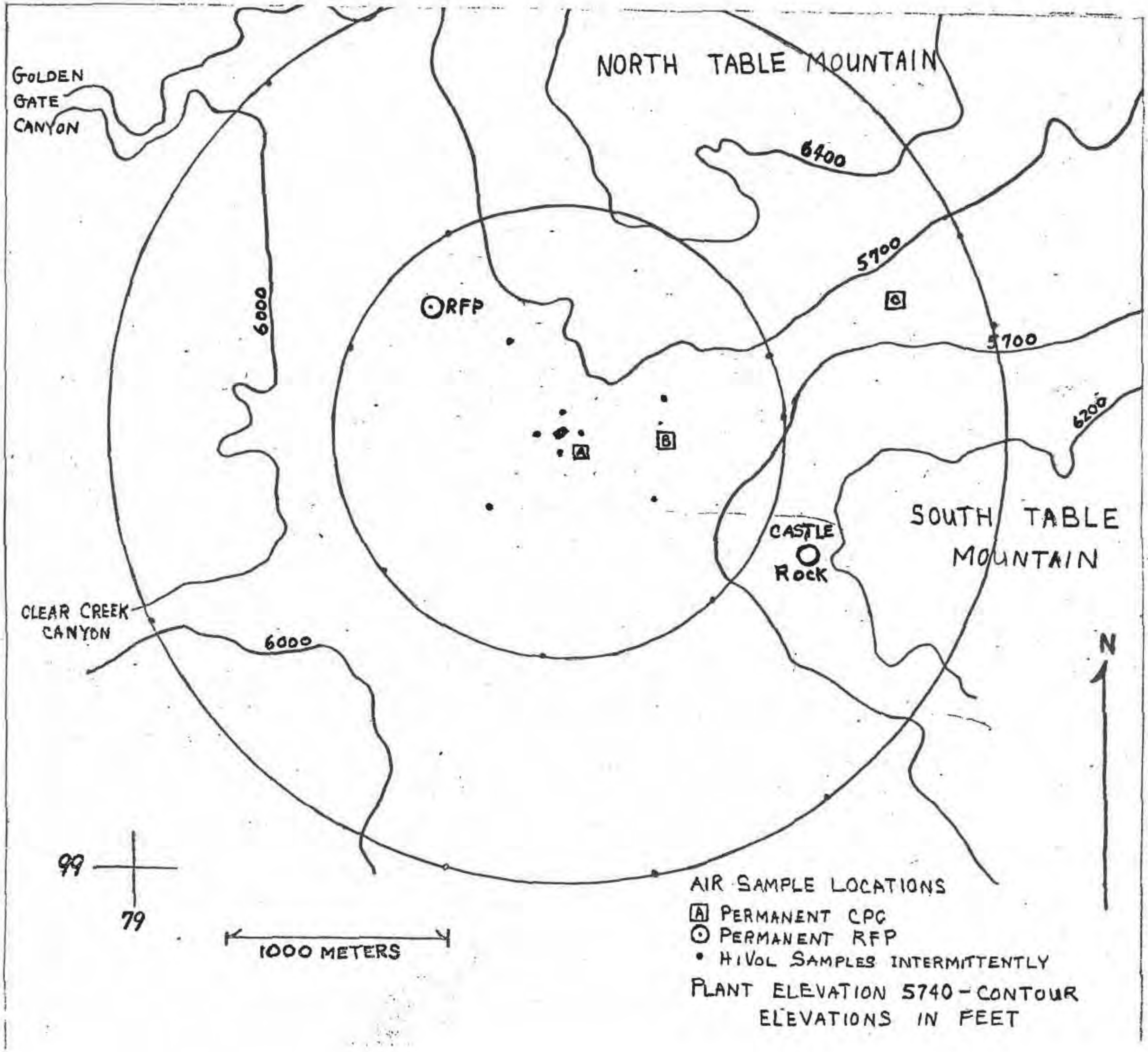
The coordinates were scaled from the U. S. Geologic Survey Maps of the area and are reported to the nearest 10 meters east and north of the grid corner. To properly locate points using this grid system, it is necessary to have a USGS map of this area with the UTM 1000 meter grid system superimposed. Referenced coordinates can then be scaled from the 1000 meter grid lines and the point identified by its location relative to surrounding landmarks.

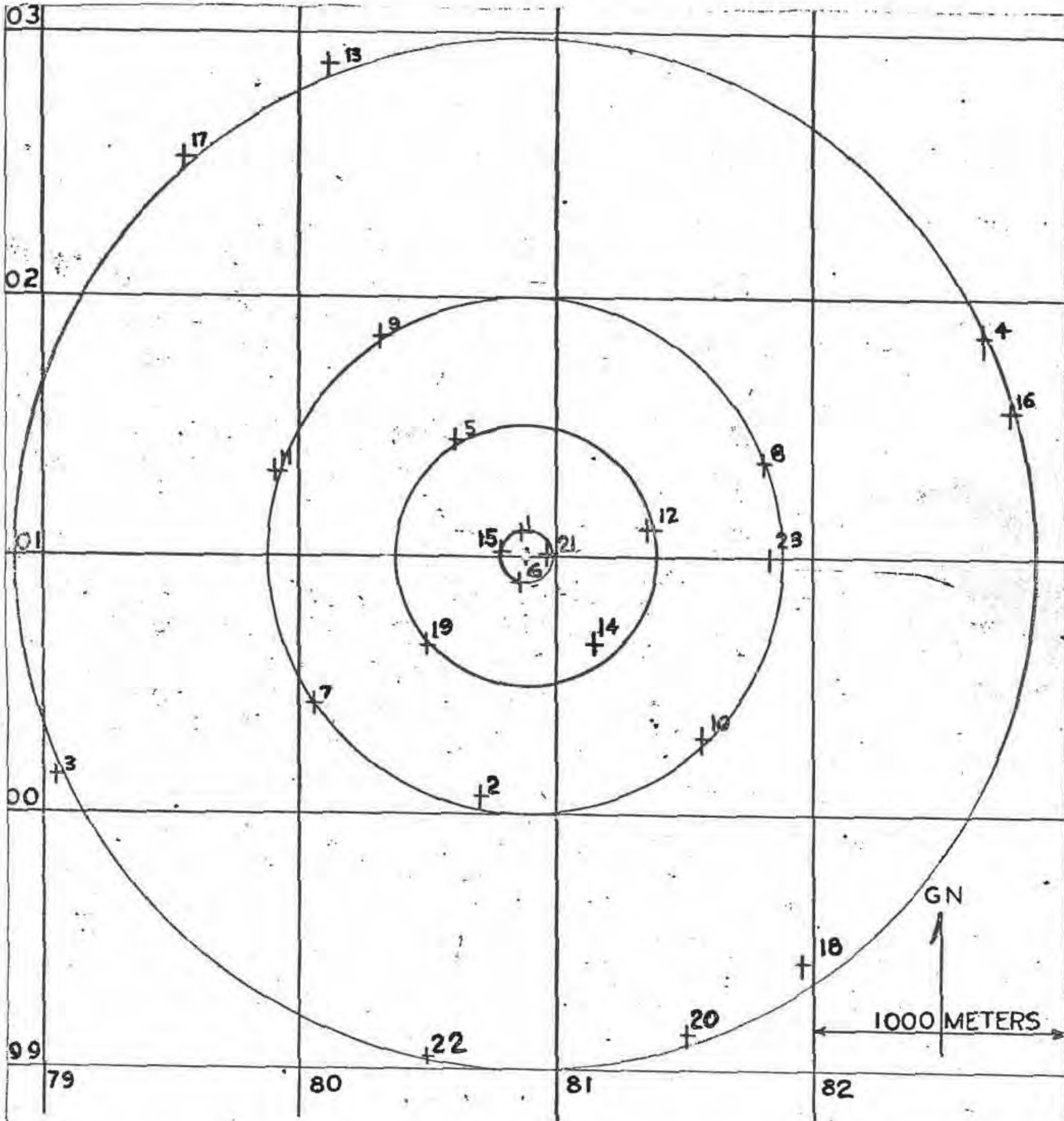
For example, sample point 4, coordinates 8267, 0185 is located 670 meters east and 850 meters north of the corner of grid square 82-01.

This system insures relocation of the sample point.



WATER SAMPLE LOCATIONS


NORTH TABLE MOUNTAIN
GOLDEN GATE CANYON
6400
5700
5700
6000
6200
RFP
A
B
C
CASTLE Rock
SOUTH TABLE MOUNTAIN
CLEAR CREEK CANYON
6000
N
99
79
1000 METERS
AIR SAMPLE LOCATIONS
A PERMANENT CPC
⊙ PERMANENT RFP
• HiVOL SAMPLES INTERMITTENTLY
PLANT ELEVATION 5740 - CONTOUR ELEVATIONS IN FEET



Point	Coordinate
1	8086 0110
2	8070 0007
3	7905 0015
4	8267 0185
5	8061 0145
6	8085 0090
7	8006 0042
8	8180 0136
9	8032 0185
10	8156 0030
11	7990 0133
12	8135 0111
13	8011 0288
14	8115 0065
15	8078 0102
16	8278 0156
17	7955 0253
18	8195 9942
19	8050 0066
20	8150 9914
21	8096 0102
22	8050 9905
23	8183 0100
FEB	8088 0100

4.2.1.3.6 **RESPIRATORY PROTECTION**

Half Mask Respirators

The Willson respirator, model 809 is equipped with two model R520 filters and is used when airborne contamination is suspected or known to exist in a particular area. This respirator provides effective filtration of toxic aerosols and extremely fine particles, including radioactive dust, and BeO particulates.

When required, the half mask respirator is used in the following manner:

Check to be sure the mask contains 2 filter cartridges securely in place.

Place the mask to your face, molding the rubber nose piece to the contour of the face.

Secure head straps.

Place palms over the two filter air inlets securely, and attempt to breathe. If mask is correctly in place it will not be possible to inhale.

After use, the respirators must be collected, washed, disinfected, inspected and reassembled for use. They are to be placed in a sealed paper bag and distributed in strategic locations.

Self Contained Breathing Apparatus

The Scott Air-Pak (Model 6000-A2 MSP) is provided for emergency use where a worker might be exposed to an environmental concentration exceeding ten times the MPC, not to exceed a period of thirty minutes of heavy exertion. The equipment is to be used in the following manner.

Connect breast strap buckle, leaving the side strap unhooked, then swing the harness over the head, snap the side strap in place, fasten the waist belt and adjust the harness to suit the user.

Check regulator hose nut at cylinder valve for tightness, open cylinder valve three full turns, snap safety chain into valve hand wheel. Check pressure indicated on regulator gauge. Adjust all of the head harness straps on the mask assembly full out. Put mask on chin first, adjust the chin straps, then temple and forehead straps last. Test the mask and hose for leaks and tightness to the face by closing the mask hose quick-connect fitting with the thumb and inhaling. The mask should collapse to the face with no air leakage. Test the exhalation valve by exhaling with the hose still closed off. During normal operation, the red knob should be fully closed, and the yellow knob fully open.

After each use, inspect equipment, scrub mask with a warm soapy solution, rinse with warm water and disinfect with a solution of 70% Ethyl Alcohol in water.

Replace used cylinder with a freshly charged one.

Store flat, preferably in a cool dry place.

4.2.1.3.7 PERSONNEL DOSIMETRY

Personal badges

The film badge program has been provided as one of the elements of a program for safe-guarding personnel against the harmful effect of ionizing radiation. It provides a periodic and permanent record of the accumulated radiation dosage. The badge is made up of a film packet designed to detect X or Beta & Gamma radiation within the 5 mrem to 550,000 mrem range and a series of silver, cadmium and aluminum foil filters to differentiate the various energy levels of radiation such as hard and soft Gamma or X-rays. The film badges are worn by all employees and visitors within the confines of the building.

Threshold Detectors

Several modifications to the standard personal film badges are made for use as area survey badges.

These badges are primarily constructed to monitor nuclear excursions, in which a high level neutron flux would be present. For this purpose an additional neutron film has been provided plus an additional series of gold, cadmium covered gold, indium and aluminum foils. These materials when irradiated with neutrons become radioactive. The degree of induced radioactivity is a function of the energy and intensity of the neutron irradiation.

The threshold detectors are located throughout the plant st strategic points. The following charts show these locations.

Films are changed monthly at which time the badges are closely inspected for damage and loose metal foils are replaced. An electromagnetic opener is provided for changing the film. The film packets are packed in a plastic bag before being placed in the shipping container, to insure un-damaged arrival. A letter is enclosed listing all terminations and new assignments during the previous month. All exposure results are kept on file and as each employee terminates, he is given an Exposure Summary, listing his external radiation exposure during his term of employment.

Urinalysis

Urinalysis is used to determine the initial background uranium content in the system and to have a periodic check on any additional uranium in the system. The frequency of collection during employment depends on the individuals work area and contact with the enriched uranium. Urine samples are collected during the first week of employment, (prior to any possible exposure), and at 3 months, 6 months, or 12 month intervals and at termination.

Two 32-ounce wide mouth "Nalgene" polyethylene bottles are packaged in a standard cardboard container and accompanied by the standard instruction form(attached). Routinely, sample bottles are distributed on Monday and collected on Wednesday. Thursday morning the samples are prepared for shipment by adding concentrated nitric acid as a preservative for each quart of sample, or fraction thereof.

Bottles are resealed and arrangments made for shipment.

A letter listing all names and sample numbers is enclosed in the shipment. A file card is maintained on each employee, listing date of sample and results. A log book is also kept, so it is possible to find the desired information either by name or by sample number. Bioassay information is also included in the employee's Exposure Summary which he receives upon termination.

4.2.1.3.8 VENTILATION SURVEYS

Instrument Calibration

The following provedure is used for calibrating the Anemotherm air meter:

Plug in air probe in 7-hole socket.

Place probe in air shield.

Depress "Velocity and S.P." button.

Depress "0-100" fpm selector button.

Depress and hold "Check" button.

Adjust indicator to red line on 0-100 fpm scale by use of the "Velocity-S.P." current adjustment knob.

Repeat similarly for 100-1000 and 1000-8000 fpm scales.

Measuring Velocity at Fixed Opening Hoods

After calibrating the instrument, locate the probe directly in the hood opening. Select the most likely scale and place the probe in the line of air movement with the red indicator dot facing the air source. Read the flow from the appropriate scale. Make necessary adjustments with the duct damper to achieve the desired velocity through the hood opening.

Measuring Velocity at Variable Opening Hoods

After calibrating the instrument, locate the probe directly in the hood opening. Operation of the instrument to determine flow rates is the same as in the preceding paragraph. Velocity measurements are made by varying the size of the hood opening to determine the maximum size of hood opening which will still produce the desired minimum velocity. The size of the opening must not be less than that normally used during operation of the hood. Necessary adjustments are made in the duct damper to achieve the desired velocity.

Static Pressure Measurements

Static pressure differentials across exhaust filters are measured at the main exhaust plenum, the P-30 filters, the mix line filters, the laboratory hood filters, and the laboratory dry box filters.

Equipment and its Use

Either the 0-2 or 0-6 inches of H_2O Magnehelic static pressure gauge may be used. Each gauge has two tygon tubes, a yellow one for use on the higher pressure or atmospheric side.

All static pressure measurement locations have two nipples, one of which is yellow.

Connect both tubes to the nipples matching yellow for yellow. Hold gauge upright and take reading.

Duct Surveys

The measurement locations are located strategically throughout the plant. Ducts are labeled A through I.

The Anemotherm Air Meter used in this operation is equipped with a ruled 36" probe extension for greater reach.

The air meter is operated as previously described and the probe inserted a specific distance to achieve a uniform air velocity profile of the duct. Six readings are taken horizontally and 6 vertically at different distances through the access holes.

Collection, Processing and Reading

Velocity measurements of all glove boxes are taken and recorded on a standard form and remarks made as to unusual conditions and damper positions. Static pressure measurements are recorded on a standard form accompanied by location, date and remarks.

Hood opening data, date taken, location and all remarks are noted on a standard form.

Duct survey data is recorded on standard forms, indicating insertion distances (R_1, R_2, R_3, etc.) sample points, (A, B, C, etc), duct diameters and duct areas. Duct survey data is processed in the following manner:

Average the 12 readings to find the average velocity in fpm.

Calculate flow rate as follows:

$$\text{average velocity (fpm)} \times \text{duct area } (ft^2) = \text{flowrate } \frac{(ft^3)}{min}$$

Correct flow rate to this altitude by multiplying by 1.275.

The rated values are:

Area	Flow Rate
I	1700 fpm
II	1000 fpm
III	3800 fpm
IV	4760 fpm
V	1950 fpm
VI	3550 fpm
VII	540 fpm
VIII	7850 fpm
IX	400 fpm

The actual values are calculated as follows:

I	=	reading at A in cfm
II	=	B-A
III	=	C-B
IV	=	D

V = E-D
VI = F-E
VII = G (I + II + III + IV + V + VI)
VIII = H
IX = I
Total to Fan = G + H + I

The entire ventilation survey is reported weekly so correct ventilation requirement and conditions can be maintained. Copies of reports are kept as a permanent record.

4.2.1.3.9 SPECIAL HAZARDS EVALUATION

Carbon Monoxide

The "Bacharach Monoxor" carbon monoxide indicator model CDE is used where carbon monoxide fumes are suspected or known to be present.

The following steps should be followed for successful operation of this device:

Unfold scale frame and scale from bottom of indicator sampler and swivel forward until it locks in place.

Slide metal tip breaker on the bottom of the indicator sampler back to open the tube breaker hole.

Insert indicator tube tip in exposed hole and break off tips to open both ends of tube. Push slide forward to close tip breaker hole.

Slide either open end of indicator tube through wire loop under scale and insert tube tip snugly into rubber tube connector of indicator sampler.

Depress push button to bottom of travel and hold down for several seconds. Release push button quickly. Allow push button to return to its original position. This will be shown by appearance of red line on push button.

When red line on pushbutton is visible, wait 15 seconds before examining the indicator tube for brownish stain in yellow colored gel. If stain appears, carbon monoxide is present in test area.

Move scale right until only unstained yellow colored gel appears in the window (A) to the left of scale's zero bar (B) and stained gel appears on the other side. Read scale at junction (c) of stained gel and white guard gel.

Carbon monoxide present is reported as % CO in the air or as ppm CO. A permanent record is kept of any unusual carbon monoxide concentrations. Area supervisors are notified when carbon monoxide is found in the area.

Hydrogen Gas

The MSA Explosimeter (model 3) is used to quickly and conveniently test an atmosphere for concentrations of flammable gases and vapors.

The following procedure should be used:

Lift the left end of the rheostat knob "on-off" bar and turn the rheostat knob one quarter turn clockwise.

Flush fresh air through the explosimeter.

Adjust rheostat knob until meter pointer rests at zero.

Place end of sampling line at, or transport the explosimeter to the point where the sample is to be taken.

Read just meter pointer to zero if necessary.

Aspirate sample through explosimeter until highest reading is obtained.

To turn off instrument rotate rheostat knob counter-clockwise until arrow on knob points to "off" position.

Report hydrogen concentration as a percent of the lower explosive limit.

Ordinarily the explosimeter is primarily used as a leak detection device and any irregularities are brought to the attention of the responsible supervisor.

A permanent record is kept of any abnormal amounts of explosive vapor found.

Microwave radiation

The "Ramcor" model 1200 Densiometer is used in the following manner:

Before connecting the antenna, set the "on-off" switch to the "off" position. Set the selector switch to "Bat" position and press the red button. The meter needle should read at or above the red dot, if it is below the red dot, replace batteries.

Select the proper antenna to operate in the frequency range to be measured. Set the "Selector Switch" to the proper band, turn the "on-off" switch to "on" and balance the meter with the balance control. Proper balance is obtained when the needle is at the extreme left of the meter.

Caution: Do not turn the instrument "on" when the selector switch is in the battery check position.

Connect the antenna to the meter and scan the desired area. The meter is calibrated in "DB" above and below 10 MW/CM^2. The 10 MW/CM^2 is the "0" db mark at the center of the scale.

If using one of the horn antennas and the polarization of the transmitting source is unknown, readings should be taken by rotating the unit until the "E" field is horizontal.

The instrument must be in "off" position when not in use.

Caution: Do not use the antenna head as a handle for removing it from the meter.

Microwave density is reported in db.

A permanent record is filed of any excess microwave radiation, and a report submitted to the supervisor to correct the situation.

4.2.1.3.10 **SHIPMENT OF RADIOACTIVE AND TOXIC MATERIALS**

REGULATIONS COVERING THE TRANSPORTATION OF URANIUM AND BERYLLIUM COMPOUNDS PERTAINING TO THE COORS PORCELAIN COMPANY.

SHIPPING PROCEDURES FOR BERYLLIUM OXIDE

General Information

This procedure is intended as a guide for the preparation of shipments of Beryllium Oxide by common carrier, parcel post, express, or courier.

Packaging

All Beryllium Oxide material will be packaged in double internal containers which will completely contain the contents under all normal shipping conditions. This double container may consist of two plastic bags separately sealed, or a combination of the following types of containers:

1. Metal can
2. Plastic bottle
3. Plastic carton
4. Cardboard carton
5. Plastic bag

Both containers shall be sealed to prevent leakage of the contents. The choice of containers will be dictated by the characteristics of the contents.

Sharp or heavy objects should be put in rigid containers, liquids in plastic bottles, etc.

The double inner container will be packed in a suitable outer container with sufficient packing material to adequately protect the contents from damage.

For packaging liquids the outer container shall be of a material chemically resistant to the liquid and sufficient absorbent packing material will be placed between the inner and outer containers to absorb the liquid in the event that the inner container should break or leak for any reason.

For packaging solids or semi-solids, the outer container shall be rigid enough to withstand the type of handling encountered in normal shipment.

Labeling

Each double inner container shall be plainly marked with a tag, label, sticker, or a felt pen in the event that none of the other three are available, with the word BERYLLIUM in such a manner as to be visible to the recipient before the double inner container is opened.

The outer container will bear in a conspicuous place the I.C.C. Class B poison label shown below:

Monitoring

Each package containing beryllium shall be monitored by Health Physics before shipping. The maximum allowable contamination level shall be 0.01 ug/cm^2 Be removable by smearing.

SHIPPING PROCEDURES FOR URANIUM COMPOUNDS IN COMPLIANCE WITH I.C.C. REGULATIONS.

General Information

As stated by the I.C.C. Bureau of Explosives, radioactive materials are a Class D poison. Radioactive material is defined as any material or combination of materials that spontaneously emit ionizing radiation. The I.C.C. has divided radioactive materials into three groups according to the type of radiation emitted at <u>any time</u> during the transportation of such a material, as follows:

Group I	Radioactive materials that emit gamma radiation only or both gamma radiation and electrically charged corpuscular rays, better known as alpha and beta radiation.
Group II	Radioactive materials that emit neutrons and either/or both types of radiation characteristic of Group I materials.
Group III	Radioactive materials that emit electrically charged corpuscular rays only, i.e. alpha or beta radiation, or any other material that is so shielded that the gamma radiation at the surface of the package does not exceed 10 milliroentgens for 24 hours at any time during the transportation of said material.

The radioactive material used and shipped by the Coors Porcelain Company, Fuel Element Department is principally an alpha emitter. Hence it is classified a Class D, Group III poison and must bear an appropriate label stating such. See Section 73.391 of the I.C.C. Regulations for specific details.

Packaging

It is necessary to exercise special care in packaging Uranium compounds. Below are listed in brief form the I.C.C. Regulations pertaining to our product.

The design and preparation of the package must be such that there will be no significant radioactive surface contamination of any kind on the outside of the container. See Section II D for maximum acceptable levels of radiation.

The smallest dimension of any outside shipping container for radioactive materials must be not less than 4 inches. Thus any package containing U-235 in any form must be at least 4 inches x 4 inches x 4 inches.

All outside shipping containers must be of such design that the gamma radiation will not exceed 200 mr/hr or equivalent at any point of readily accessable surface.

The outside shipping container for any radioactive material other than exempt quantities, (See Section II E for exempt quantities) shall be wooden boxes, fiberboard boxes, fiber drums or metal drums. Any container used for shipping radioactive materials must be constructed so as to withstand conditions incidental to shipping.

Radioactive materials Group I liquid or solid must be packaged in such a manner that no alpha nor beta radiation is allowed to escape to the exterior surface of the outside container and the gamma radiation may not exceed ten milliroentgens at one meter from any surface of the radioactive source.

Liquid radioactive materials must be packed in tight containers, chemically compatible with the contents and must be surrounded with sufficient absorbent material to entirely absorb the contents in the event the container should leak for any reason. Absorbent packing material must be such that its efficiency is not impaired by chemical reaction with the contents of the container.

Labeling

Outside containers containing Group I or II radioactive materials must bear an appropriate red label as described in paragraph 73.414 a of the I.C.C. Regulations.

For rail, truck, or express shipment

For air shipments

The radioactive materials shipped from Coors Porcelain Company, Fuel Element Department being a Class D Group III poison must bear a 4 in. x 4 in. <u>blue</u> and <u>white</u> label.

For rail, truck, or express shipment

For air shipments

Inner containers shall bear a sticker, label, or tag in such a place that it will be easily noticed by persons opening these containers. See Section 73.393 and 73.414 of the I.C.C. Regulations for specific details.

It is unlawful to label packages with Class D poison labels if the packages do not contain such. It is also unlawful to leave a Class D poison label affixed to any vehicle after it has been unloaded of radioactive contents or to ship emptied containers with Class D poison label affixed.

All radioactive material shipments which contain Beryllium Oxide will display the Blass B poison label in addition to the appropriate Class D label.

Monitoring

In brief the following conditions must be met before a shipment of radioactive material may safely leave the Fuel Element Department:

There may not be more than 50 dpm surface contamination removable by smearing on any outer or inner container.

Gamma radiation may not exceed 200 milliroentgens per hour at any readily accessible surface nor may it exceed 10 mr/hr at one meter from any point on the radioactive source.

Packages must be put together in such a manner that their contents will not spill or in any way cause undue exposure to personnel involved in handling the shipment. Packages must be able to withstand conditions incidental to shipping.

All packages must bear in a conspicuous place appropriate labels denoting the nature of the contents.

The Health Physics Department will monitor packaging procedures periodically to make certain that all rules are being followed.

Exempt Quantities

The I.C.C. Regulations allow for unmarked shipments providing that the quantity being shipped meets the following requirements:

The package must be such that there can be no leakage of the contents under conditions normally incident to transportation.

The package may not contain more than 19 grams of Enriched Uranium or 2000 grams of Natural Uranium.

Surface contamination may not exceed 50 dpm/ft^2 removable by smearing and gamma radiation from within the package may not exceed 10 milliroentgens per 24 hours at any readily accessible surface.

Shipments of low level waste may be made in car load lots providing gamma radiation does not exceed 10 milliroentgens per hour 12 ft from any surface or 5 ft from either end. Material must be packed in strong tight containers braced in the vehicle with no loose material in the vehicle. Shipments must be loaded by consignor and unloaded by consignee and the vehicle must bear a 10-3/4 in. x 10-3/4 inc. label as described in Section 74.553 of the I.C.C. Regulations. See Section 73.392 of the I.C.C. Regulations for specific details.

POSTAL REGULATIONS COVERING THE TRANSPORTATION OF RADIOACTIVE MATERIALS

General

Quantities that may be shipped via Parcel Post are those that may be shipped via R.E.A. or common carrier as exempt quantities. There are some differences however in the labeling requirements. Below are listed the requirements in brief.

Packaging

Packages must be put together as outlined in Section II, B, 6 of this paper and must be able to withstand conditions normal to shipment via Parcel Post. No shipment may contain more than 19 grams of Enriched Uranium or 2000 grams of Natural Uranium.

Labeling

Packages must be labeled as follows:

> "Radioactive material, gamma radiation at surface of parcel less than 10 mr/24 hr. No significant alpha, beta or neutron radiation".

The contents must comply with the specifications designated on the label.

Monitoring

Monitoring will be carried out as outlined in Section II D, 1, 3 and 4.

Exempt Quantities

There are no exempt quantities for shipment via Parcel Post. Any and all packages containing radioactive materials must be labeled.

REGULATIONS COVERING AIR SHIPMENTS

Shipments made by air must meet all requirements of the I.C.C. plus those listed below.

Packages must bear a label reading as follows:

> "Do not place in same compartment with undeveloped film or mail".

No more than 40 units of Group I and II materials may be shipped on one air craft. This will necessitate marking the contents of each package providing the total shipment contains more than 40 units (1 unit-1 mr/hr at one meter distance of hard gamma).

Compensation should be made for differences in air pressure during flight, i.e. caps on air tight containers should be secured to withstand pressure from within the container.

4.2.1.3.11 MEDICAL PROGRAM

Schedules

A pre-employment physical examination is required of all employees. Routine physical examinations are scheduled for either 6 months or 12 months intervals, depending upon the person's work area or job duties. A terminal physical examination is required of all employees just prior to termination. All employees are checked for weight change, vital capacity change, any chest, throat, skin complaints on either a monthly, bi-monthly or tri-monthly basis. The same system, to determine frequency of visits, is used for the routine medical visits and the scheduling of routine physical examinations.

Any employee with a suspected exposure or occupational illness is sent immediately to Dr. Wright for examination. Any severe change in weight, or vital capacity is brought to the Doctor's attention so that he may evaluate the situation and decide what action should be taken.

Several of the A. Coors construction workers are included in the medical program, since their job duties in this division bring them into close contact with possible contamination.

All employees are required to have ten weekly checks of weight and vital capacity, which are used as a background, to determine any abnormal change.

A Medical Record form is kept on all employees, listing their starting date, their pre-employment physical date, any routine physical examinations and routine medical visits. A file card system also lists weight and vital capacity and the dates of routine medical visits for each employee. A list of physical examinations and routine medical visits is included in the Exposure Summary which each employee receives upon termination.

A copy of the Maternity Leave form is attached. This form, when signed, is placed in the employee's file in the Personnel Department.

Maternity Leave Form to be Signed by All Female Employees:

TO: All Female Employees

SUBJECT: Maternity Leave

The Company policy regarding maternity leave for women in the Fuel Element Department is as follows:

1. It is the employee's responsibility to notify the Company at the first knowledge of pregnancy and to request maternity leave at that time.

2. Maternity leave will be granted immediately for a period not to exceed two months beyond the termination of pregnancy.

To be signed by the employee:

I understand the foregoing statement of policy and agree to assume the responsibility for notifying the Company at the earliest possible knowledge of pregnancy and to accept immediate maternity leave.

Date	Employee's signature

4.2.1.3.12 PERSONNEL INDOCTRINATION

Health Physics Lectures

The Health Physicist meets with all new employees to discuss the health physics safety program. The contents of the Health Physics Guide are covered with the new employee. Special emphasis is placed on proper usage of respirators, contamination enclosures, and protective clothing.

Nuclear Safety Lectures

The Nuclear Physicist meets with each new employee who will be working in uranium processing areas. The major provisions in the Nuclear Safety Guide, CPC-68 are covered with the new employee. The emergency evacuation procedure is also covered in detail.

Films

The film "Living with Radiation" is shown to all new employees. The film "Criticality" is shown to all new employees who will be handling uranium.

On the Job Training

It is the primary responsibility of a supervisor to train a new employee for his job. This training is to include the health and nuclear safety aspects of a particular job.

Nuclear Safety Inspectors and Health Physics Technicians have the responsibility to monitor all production areas to see that all aspects of safety are complying with rules and regulations.

4.2.1.3.13 DECONTAMINATION

Area Decontamination

In the instance of a spill involving hazardous materials, Health Physics should be notified immediately.

They will determine the type and extent of contamination and the most suitable method of decontamination. In the event of any possible airborne contamination, personnel should wear respirators during clean-up.

Care should be taken to minimize any spreading of the contamination by taping off the area, thus cutting down on traffic.

Depending on the degree of contamination, equipment may either be cleaned in place, or removed to the Decon Room and cleaned within either hood.

Personnel

Protective clothing has been provided to eliminate, as much as possible, the contamination of personnel. In some cases, however, an operator may be unavoidably contaminated. He must then wash thoroughly with soap and water and when necessary with a more effective decontamination materials.

Decontamination Materials

Decontamination materials are listed approximately in order of effectiveness.

Alconox and water.

Alcohol or Acetone for greasy or oily items.

Decon Solution which is prepared as follows:

Dissolve 227 g of Citric Acid and 113 g Versene Powder in water in a 1500 ml beaker.
Add 37 ml of Igepal.
Pour concentrate into a 5 gal. container and dilute to 5 gallons.

Nitric Acid, either diluted or concentrated, depending on the requirements.

When contamination cannot be removed with these materials it may be painted over.

4.2.1.3.14 NUCLEAR SAFETY MONITORING

Operating Principles

Uranium in geometrically unsafe containers is handled in 300 g batches.

Uranium in geometrically safe containers is limited by the volume of the container and the hydrogen to uranium ratio.

Minimum edge to edge separation of twelve inches is maintained between batches of uranium during all stages of processing, storage, and transportation, unless safe geometry controls are used.

Stored batches are maintained in verticle planar arrays to minimize interaction.

Operating criticality limits are posted wherever uranium may be handled.

The nuclear safety plan is contained in the Nuclear Safety Guide, CPC-68, Rev. 1.

Inspection of Work and Storage Areas

Nuclear Safety Inspectors continually monitor work on all shifts.

Visual inspections are made to insure that uranium is handled only at approved work stations and storage areas, in accord with the operating principles.

Inspections are made to verify that the uranium being processed at a station, does not exceed the station criticality limit.

Gamma surveys are made to insure that uranium does not accumulate in unsafe places such as low level waste tanks, BeO storage areas, etc.

Infractions

A nuclear safety infraction is any action which is not explicitly permitted in the Nuclear Safety Guide, CPC-68.

All infractions are reported on the Nuclear Safety Infraction Report by the Nuclear Safety Inspector monitoring the area where the incident occurs.

The Nuclear Safety Committee studies the report to determine what disciplinary action, if any, is to be taken against the violator. Disciplinary action is then administered by the violator's supervisor.

Individuals violating provisions of the Nuclear Safety Guide are subject to the following disciplinary action:

A. Personal contact by supervisor and Nuclear Safety Committee to include a written reprimand.

B. Personal contact by supervisor and Nuclear Safety Committee to include a disciplinary layoff of from one to five days.

C. Personal contact by supervisor, Division Manager, and Nuclear Safety Committee for purpose of immediate discharge.

For a first offense, the Nuclear Safety Committee will review the situation and recommend action A, B, or C, depending on the circumstances of the offense.

For a second offense, the Nuclear Safety Committee will review the situation and recommend action B or C, depending on the circumstances of the offense.

For a third offense, action C is the only permissible action.

Training of Operators

All operators working in areas where uranium is handled must be given a nuclear safety indoctrination by the nuclear safety group and by the operators' supervisor.

The operators' actions are closely watched by the Nuclear Safety Inspectors until the operator demonstrates he understands the importance of following nuclear safety rules.

Date______________________ Violator (______________________

Time______________________

Location______________________

A. Detailed Explanation of Situation:______________

B. Nuclear Safety Limits which were in effect:

C. Remedial Action taken and by whom:

____
Nuclear Safety Inspector

Nuclear Safety Committee Recommendation: Date______________________

Corrective action and by whom: Date______________________

Changes in Operating Procedures

Changes in the nuclear safety plan are requested on the Nuclear Safety Deviation Request form.

Minor changes in the plan can be approved by three members of the Nuclear Safety Committee.

Major changes in the plan must be approved by the AEC-SAN office.

Emergency Conditions

A system of five radiation detectors is installed in the Fuel Element Building to give warning when the radiation levels within an area exceed a preset level. The most likely cause of high radiation levels is an uncontrolled chain reaction (criticality).

The alarm will be enunciated by means of a siren located at the detector. Each detector and its accompanying siren operates independently of the others.

When a siren sounds, all personnel in the building will:

> Proceed to the assigned exit for the area in which they are working.
>
> Do not stop to change clothing, shoes, badges, etc.
>
> Proceed to the designated assembly area.
>
> Report to the emergency warden at the assembly area.

Re-entry to the building will be accomplished only after Management and Health Physics have evaluated the situation and have approved re-entry.

Each work area will have an assigned exit and evacuation route which will take personnel to the most feasible exit by the safest route. The accompanying floor plans diagram the exits and the areas to use these exits.

The exits are indicated as:

A - Front Entrance to Building
B - South Side of Building, East Crash Door
C --South Side of Building, West Crash Door
D - Escape Hatch from Depressed Area
E - Furnace Room Exit
F - Mezzanine Exit

The following areas will use the exits designated below:

A - Offices, Laboratory, Locker Rooms, Vault, Mechanical Technicians Area, Tool Crib, Decontamination Room, Second and Third Levels of Tower, Blanchard and Drum Cut-off Grinding Area.

B - Ground Floor Level of Tower, Extrusion, Despatch Oven.

C - Mix Line, Grinding.

D - Depressed Area.

E - Furnace Room, Inspection.

F - Mezzanine, Alternate for Offices.

All personnel after evacuating will assembly in specified areas according to their normal work areas. The Warden for each area will determine if all of the people from his area are present or accounted for. The Area Warden will report to the Chief Warden when this determination has been made.

The Chief Warden, the Health Physicist, and designated Supervision will assemble at the main entrance to the Fuel Element Building.

Personnel will evacuate to the Porcelain Plant by the following route:

Proceed directly to the small door on the north side of the main plant and enter.

Inside the building, proceed south between the two large kilns to the large metal doors opposite the stairway. Turn right and proceed west to the courtyard.

Upon reaching the courtyard turn right and enter the raw material storage area. This area consists of a long hallway with a series of vaults opening on to the hallway.

Assemble in the hallway in groups according to your normal work area. Signs will be posted in the hallway designating assembly areas.

The assembly areas will be designated from west to east in the following order:

A - Offices
B - Laboratories
C - Furnace Room, Mezzanine
D --Inspection
E - Mech Tech
F - Grinding
G - Chem Process
H - Misc., Visitors, Maintenance, Construction, Accountability, Health Physics, Nuclear Safety, etc.

Upon arriving at the secondary assembly area, the Area Wardens will take charge of their groups and await further instructions.

It is absolutely essential that all personnel involved in an emergency evacuation observe the following rules:

Proceed immediately to your assigned assembly area by means of the designated exit.

Report to your Area Warden immediately.

Do not leave the assembly area until instructed by your Area Warden.

<u>Do not discuss with anyone</u> any details of the emergency situation. Public statements to the Press, etc., will be made only by authorized persons.

Plant protection personnel will assist by whatever means necessary to keep out unauthorized personnel and to maintain clear traffic lanes for emergency vehicles.

Normal procedures will be followed for any emergency conditions which occur at night or on weekends.

After accounting for all personnel, the Chief Warden will then telephone the following personnel in the order listed:

Bob Mornin	424-6283
George Bidinger	424-6667
Dale Smith	443-1291
Tony Reeder	355-3721
Frank Mayer	355-8852
Joe Coors	279-4108

Plant protection personnel will maintain security surveillance from outside the building until proper authorities arrive to evaluate the situation.

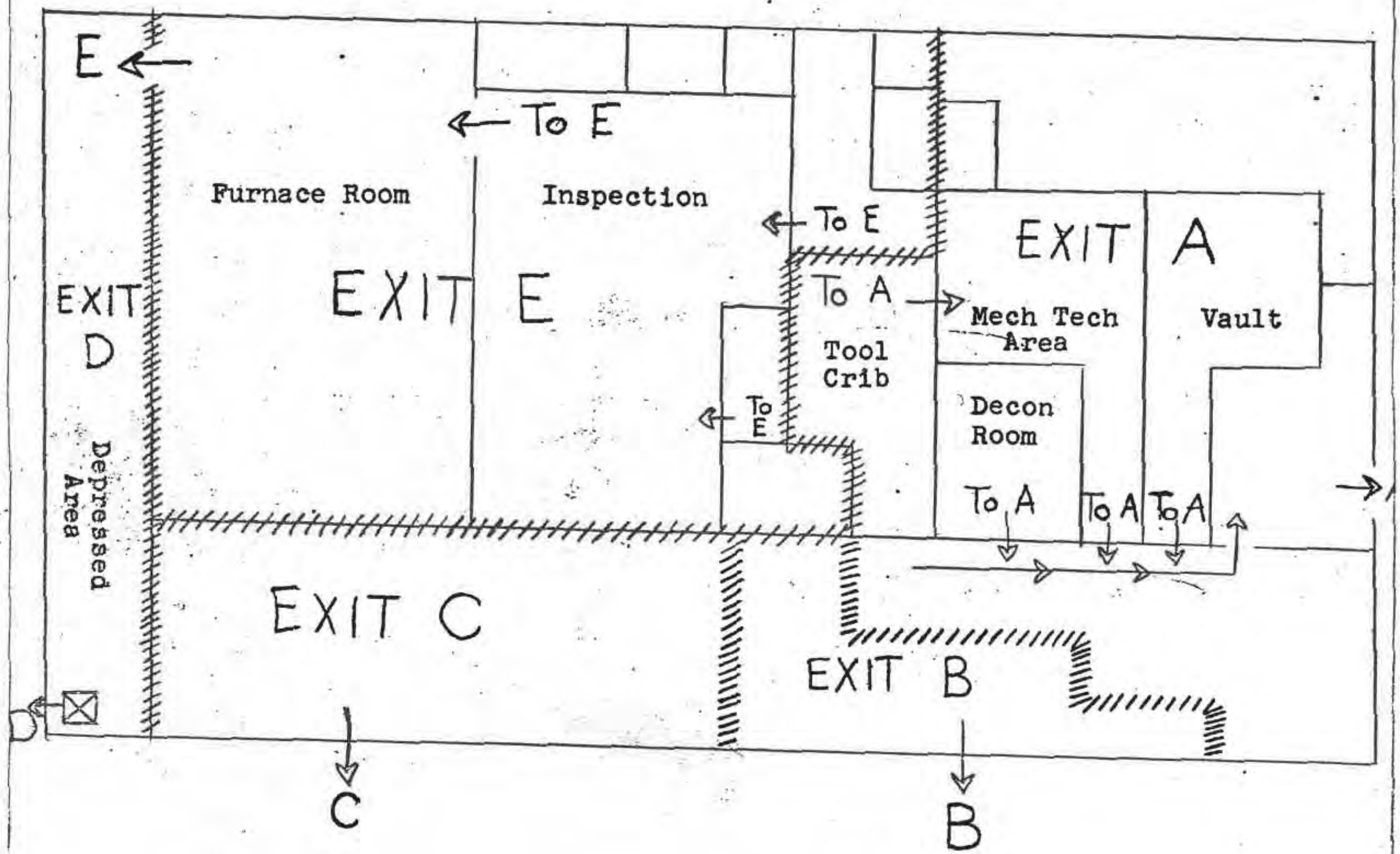
The Chief Warden will meet the above personnel at the Fuel Element Building.

The following emergency equipment will be kept at the main guard post and will be removed to the outside of the main entrance by the guard in case of emergency:

2 Scott Air Paks

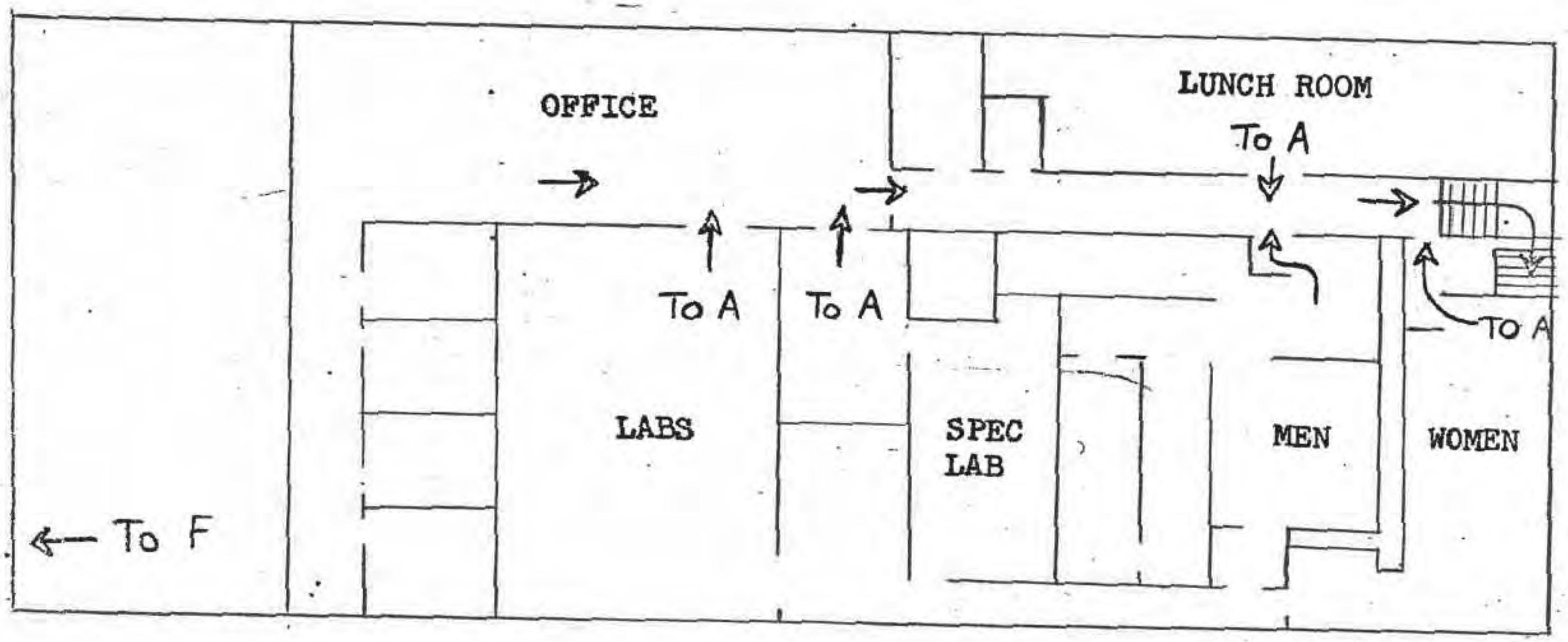
1 Emergency Kit containing:
- 2 pair coveralls
- 2 respirators
- 1 High Range Gamma Survey Meter
- 1 Alpha Survey Meter
- masking tape
- rubber gloves
- hoods
- Flashlight
- spare batteries

EVACUATION ROUTES FUEL ELEMENT BUILDING



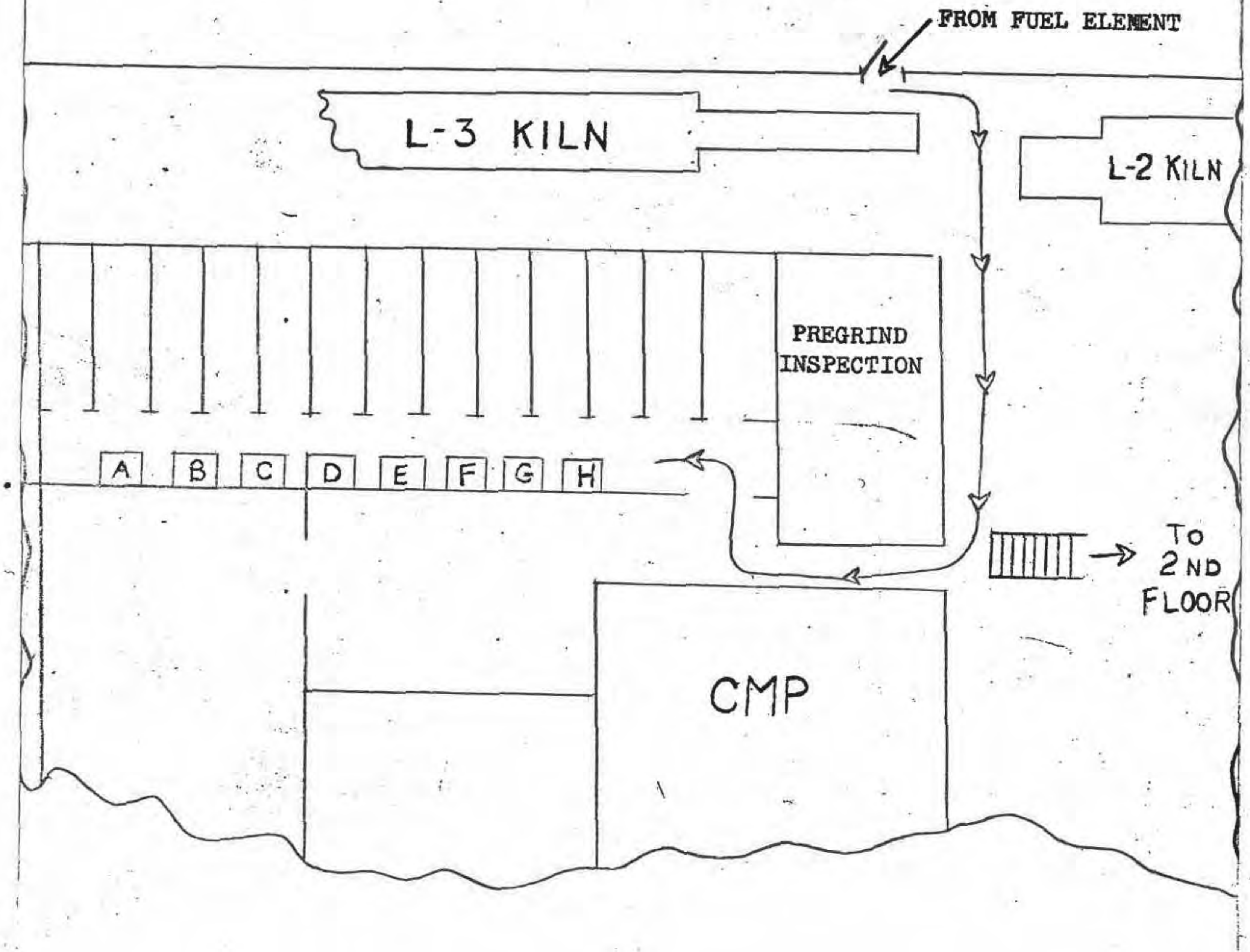
EVACUATION ROUTES, FUEL ELEMENT BUILDING

SECOND FLOOR



ALL SECOND FLOOR EVACUATES BY EXIT A, MAIN ENTRANCE

ASSEMBLY AREAS IN COORS PORCELAIN COMPANY



COORS PORCELAIN COMPANY

GOLDEN, COLORADO
Phone 279-4533

January 23, 1964

DOCKETED
USAEC
LICENSING & REGULATION
JAN 28 1964
asc
DOCKET OFFICER

RECEIVED
JAN 23 1964
U.S. ATOMIC ENERGY COMMISSION
Division of Licensing & Regulation

United States Atomic Energy Commission
Division of Licensing and Regulation
Bethesda, Maryland

Attention: Donald A. Nussbaumer, Chief
Source and Special Nuclear Materials Branch

Reference: Docket Numbers 70.814 and 40-7096 - See Rpts File for Original + Encls.

LAR File Copy

Dear Mr. Nussbaumer:

The attached submission represents the completion of our application for licenses to receive, use, process and transfer Source and Special Nuclear Materials. This submission consists of two parts. The first part is a group of revisions to our original application of December 19, 1963. These revisions are the result of a telephone conversation between Mr. Harmon of your office and our Mr. Wasson on January 15, 1964. The second part is Appendix B to the original application which outlines our Nuclear Safety plan.

In answer to Mr. Harmon's questions, please refer to the following revisions:

Personnel contamination monitoring as they leave work area	A-5
Clarification of nomenclature regarding waste systems	A-10 to A-12
Reference to fire resistant air filters	A-15 A-17 to A-19
Statements to satisfy 10 CFR 70.24(c) (Effects of fire, explosion, etc.)	A-12A App B
Description of fire protection	A-38
Maps of area	17A and 17B
Handling of pyrophoric materials	A-40
Licensing of laundry facilities	A-2

Statement of insurance coverage - (enclosed in envelope addressed to Mr. Charles Lovejoy and is to be included with our previous statement of financial qualifications to be treated as Company Confidential.)

This submission should be placed in the loose leaf binder sent to you with our application of December 1963. Please refer to the attached revision control sheet to delete obsolete pages and to include revised pages and new pages in the application as a whole.

We trust that with this material, our application for the Source and Special Nuclear Material licenses will be complete. Please advise us if any further information is required.

Very truly yours,

COORS PORCELAIN COMPANY

B. L. Mornin

B. L. Mornin, Manager
Energy Products Division

RDS:br

Attachment

January 20, 1964

Schedule of Changes, "Application for AEC License to Receive, Process, Use and Transfer Source and Special Nuclear Material.

Delete			Insert		
Page	3	12/15/63	Page	3	1/20/64
				17A	1/20/64
				17B	1/20/64
	18	12/15/63		18	1/20/64
Table of Contents			Table of Contents		
Appendix A		12/15/63	Appendix		1/20/64
	A-2	12/15/63		A-2	1/20/64
	A-5	12/15/63		A-5	1/20/64
				A-5-A	1/20/64
	A-10	12/15/63		A-10	1/20/64
	A-11	12/15/63		A-11	1/20/64
	A-12	12/15/63		A-12	1/20/64
				A-12-A	1/20/64
	A-15	12/15/63		A-15	1/20/64
	A-17	12/15/63		A-17	1/20/64
	A-18	12/15/63		A-18	1/20/64
	A-19	12/15/63		A-19	1/20/64
	A-38	12/15/63		A-38	1/20/64
				A-39	1/20/64
				A-40	1/20/64
				A-41	1/20/64
			Appendix B, pages B-1 to B-64		1/20/64

DOCKET NO. 70-814 + 40-7086

L&R File Copy

trans. w/11-14-63 HR

LICENSE APPLICATION SUPPLEMENT NO. 3
November 14, 1963

DOCKETED
USAEC
Division of
Licensing & Regulation
NOV 18 1963
Docket Officer

Health Physics Guide

This copy of the Coors Porcelain Company "Health Physics Guide" (Amended November 4, 1963) is furnished for reference use as a supplement to Coors Porcelain Company applications for AEC licenses.

RECEIVED
NOV 18 1963
Atomic Energy Commission
DIVISION OF
LICENSING & REGULATION

7666

HEALTH PHYSICS GUIDE

COORS PORCELAIN COMPANY

GOLDEN, COLORADO

Prepared by:

R. D. Smith

FEBRUARY, 1961

Amended November 4, 1963

TABLE OF CONTENTS

I. INTRODUCTION

This manual outlines the Health Physics and Industrial Hygiene problems associated with the fabrication of ceramic fuel elements. These ceramic fuel elements may contain highly enriched uranium, low enriched uranium, normal uranium, depleted uranium, thorium oxide, and beryllium oxide.

The problems associated with the accountability, criticality, and security phases of this division are discussed under separate sections. The criticality manual appears as classified document number CPC-68. The accountability manual was submitted to AEC-SAN as "SS MATERIAL CONTROL PROCEDURE MANUAL", dated October, 1960. The security manual was submitted to AEC-SAN as "COORS PORCELAIN COMPANY SECURITY PROCEDURE MANUAL", dated December, 1960.

The overall responsibility for Health and Safety rests with the Division Manager, Mr. B. L. Mornin. The direct responsibility for administering the Health and Safety program is assigned to the project Health Physicist, Mr. R. D. Smith. The Health Physicist reports directly to the Division Manager through the Production Superintendent.

Trained Health Physics representatives will be present in the processing area at all times that hazardous materials are being processed.

The basic health hazards in this operation are of a toxicological and radiological nature. Both the radioactive material and the beryllium being processed present the greatest hazard to the worker when particulates of these materials are inhaled. While the effects on the [illegible] may vary from one material to the other, the problems faced in p[illegible]tecting the worker are similar, and a control program which is effective for one material will likewise be effective for [illegible] other.

Enriched uranium presents an added hazard in that the material will enter a critical reaction if certain parameters are met, [illegible]ating great quantities of external neutron and gamma [illegible] which can seriously, if not fatally, expose a worker. The [illegible]inistrative and physical control of the enriched uranium is outlined completely in document CPC-68. This manual will deal only with the detection and measurement of the radiation associated with enriched uranium and [illegible] radioactive materials under normal and abnormal conditions.

While the relative hazards of beryllium and radioactive materials will vary with the quantities present, the toxicological hazard of beryllium will be the controlling factor in all design criteria where both materials [illegible] handled. If a design is adequate to control the hazard from be[illegible]ium, the hazard from other radioactive materials handled (exclusive of criticality) is adequately controlled.

II. CONTROL OF AIRBORNE CONTAMINATION

A. General

In controlling the potential airborne contamination problem, the philosophy of confinement and containment will be pursued in preference to reliance on individual respiratory protection. The basic equipment provided for the confinement and containment of toxic materials are the total enclosure (glove box) and the partial enclosure (velocity hoods).

B. Glove Boxes

The basic design criteria for glove box design for this operation are as follows:

1. Tightness. The enclosure will be sealed tight all around to prevent atmospheric contamination even in the event of exhaust air failure. The air brought into the enclosure is introduced through a high efficiency filter at one of the sides.

2. Cleanability. Cleanliness is important from a quality control standpoint. Ease of maintaining cleanliness is therefore important. This calls for smooth interior surfaces, rounded corners and absence of hard to reach areas.

3. Workability. The locations of glove ports, window angles, vestibules, etc., are such as to entail a minimum of fatigue to the operator.

4. Air Requirement. Air flow capability is such that there will be a flow of 200 fpm across the face of any opening to contain all contaminants within the dry box. This means that should a glove break or a vestibule door be opened there will be a surge of air through the opening of such magnitude as to have this velocity. This is provided for by maintaining a flow of 30-50 cfm @ 0.5-0.6 inches of water static pressure. The air is drawn through an absolute type filter (99.9% efficient down to 0.3 micron particle size) and exhausted through a 2-inch diameter outlet with a velocity of 500 fpm through the filter and 1000 fpm through the outlet. Capability of the exhaust blower is the sum of all of the dry box flows @ 5 inches of water static pressure.

C. Velocity Hoods

The basic design criteria for velocity hood design for this operation are as follows:

1. Hood Openings. All hood openings are kept to a minimum size commensurate with the operation requirements of the hood. In many of the enclosures doors are provided to close the openings.

2. Air Requirements. The minimum velocity maintained across the face of any hood is 150 fpm. In certain operations where large amounts of contaminants are generated, the minimum velocity will be maintained at 400 fpm. The flow rate designed for each enclosure is such that the minimum desired velocity will be maintained even if all hood openings are open simultaneously.

3. Air Filtration. Air exhausted from a velocity hood is passed through an absolute type filter (99.9% efficient down to 0.3 micron particle size) before being discharged.

III. RESPIRATORY PROTECTION

While the system of confinement and containment will provide excellent control of airborne contaminants, it must be realized that situations may arise when airborne contamination is suspected or known to exist in a work area. When this situation exists, measures must be taken to protect the workers in the area until the desired containment is again achieved. Two types of individual respiratory protection have been provided.

A. Half Mask Respirator

Each worker will be provided, when necessary because of suspected air-borne contamination, with a Willson respirator, Model 809, equipped with Model R520 filters. This filter is designed for effective filtration of toxic aerosols and for extremely fine particles, effectively filtering the finest smokes as well as highly toxic particles including radioactive dusts and beryllium particulates.

A 20% penetration has to be assumed with any half mask. (Handbook for Radiation Monitoring, LA-1835). Since it is unlikely and undesirable for individuals to wear respirators for more than 50% of any working day, the upper limit of air borne contamination for which the half mask will be work is 10 times the MPC_a.

B. Self Contained Breathing Apparatus

Scott Air-Pak, Model 6000-A2MSP, is provided for emergency use where a worker might be exposed to an environmental concentration exceeding 10 times the MPC_a. This breathing apparatus is approved for a period of 30 minutes of heavy exertion. It has the feature of maintaining a slight positive pressure within the facepiece, minimizing any penetration of the contaminated atmosphere into the facepiece due to poor fit, leaks, etc.

IV. AIR SAMPLING

A. Process Area Sampling

A comprehensive program of air sampling will be conducted on a routine basis to provide day by day evaluation of normal plant operations. Special samples will be taken to evaluate new or modified operations, maintenance work, or unusual conditions.

To assure adequate sampling of the general air, a number of portable air pumps capable of continuous operation are provided. Each of these pumps will be strategically located to provide coverage of all operations and areas. These pumps operate continuously during working hours at a flow rate of 2 cfm, the normal inspiration rate of a man at work.

The type of pump used for portable samplers will be similar to the Gast Model 0740 pump "Giraffe Sampler".

To collect breathing zone samples required by the Atomic Energy Commission recommendations, two types of sampling equipment are available. The portable Gast sample pumps can be located in the proper location with respect to the operation, air currents, and personnel. To sample operations of a short duration, the High Volume air sampler of the type manufactured by the Gelman Instrument Corporation or the Staplex Manufacturing Company will be used.

Air samples obtained with the portable pumps are collected on an ashless filter paper of a 2-inch diameter. Whatman #41 is a typical paper of this type. The flow rate will be measured by controlling the pressure drop across the filter paper, using a Magnehelic vacuum gage as the indicating instrument. Flow rate curves have been determined for this type of paper and sample holder, equating flow rate to pressure drop. Flow [illegible] sampling pump will be regulated by a gas cock type of valve.

Duct Sampling

Air being discharged from the building is sampled continuously for beryllium and uranium content. Because of the proximity to populated areas, stack effluents will be held to the recommended environmental MPC levels of 0.01 ug/M^3 for beryllium [illegible] ug/cm^3 for [illegible]. Whenever a thorium or thorium compound are being processed in the building the environmental MPC level will be 4×10^{-13} uc/cm^3 for gross alpha activity from mixed nuclides.

[illegible] air which has passed through process equipment and enclosures must pass through absolute type filters before leaving the building. Room air is controlled by exhausting through the [illegible] ventilation system.

To assure [illegible] collecting of particulates, a pitot sampling tube is introduced into the exhaust air duct drawing air at a flow rate which will produce isokinetic conditions at the pitot orifice.

C. Out-Plant Air Sampling

Off site air samples will be collected by means of two systems. Three permanently installed air sampling stations will collect air continuously on a weekly basis. The Gelman Nuclear Air Sampler will be used to collect these samples. The samples will be collected at a flow rate of 1 cfm.

High volume samples of a shorter duration will be collected at various locations around the plant environs. These samples will be taken at frequent intervals prior to plant start-up and on a continuing basis of 3-4 samples a week after plant start-up.

The location of the three permanent sampling stations is in the vicinity of the squares marked A, B, and C on the accompanying diagram. The UTM grid coordinates of these stations are:

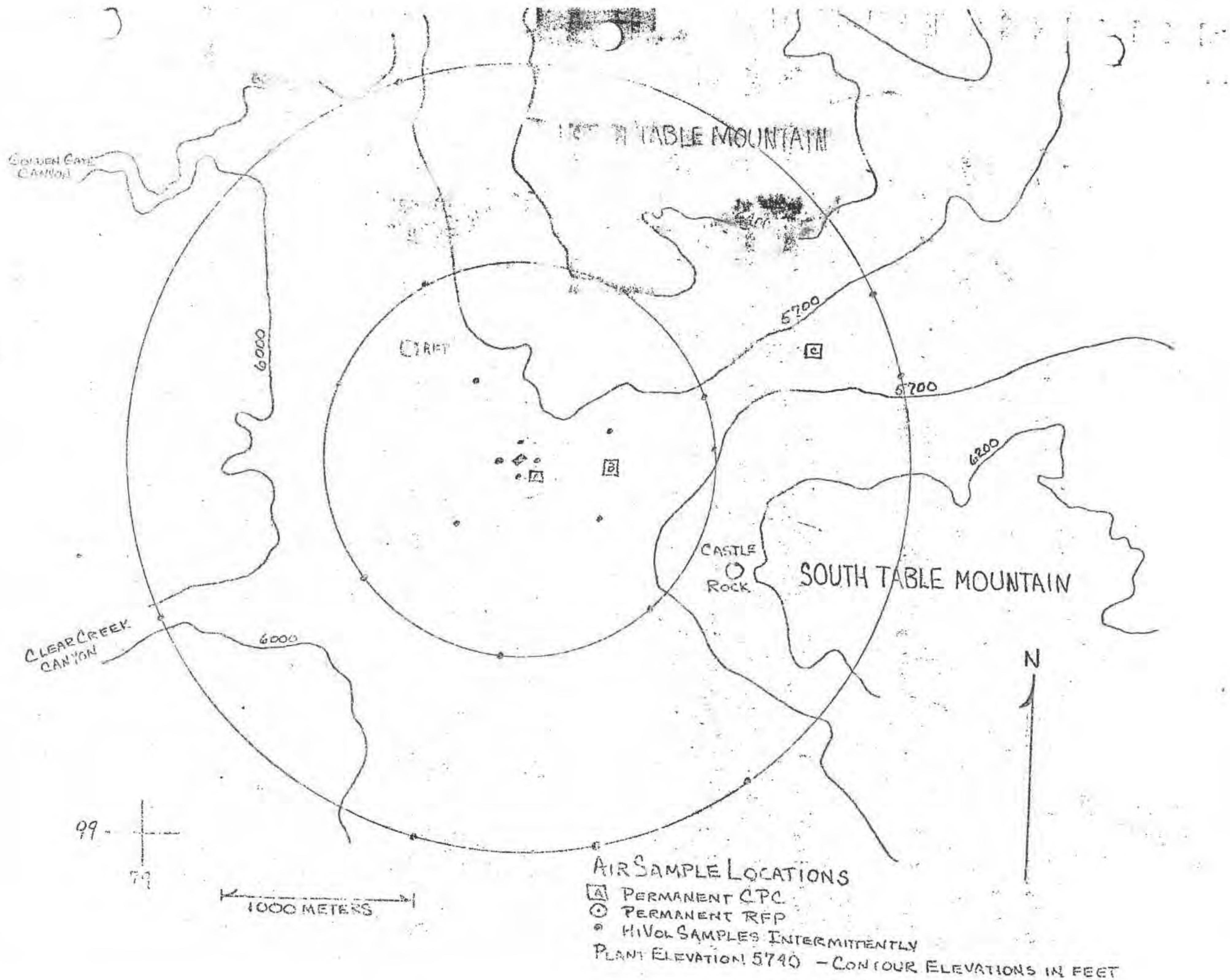
A. 8092 0095. 100 meters southeast of plant.

B. 8135 0100. 500 meters east of plant.

C. 8240 0165. 1600 meters east-northeast of plant.

These stations will be supplemented by a permanent sampling station operated by the Dow Chemical Company. This station is located at coordinates 8023 0158, approximately 850 meters northwest of the plant.

In the event that the maximum average neighborhood concentration at the ground during any calendar month,


GOLDEN GATE CANYON
TABLE MOUNTAIN
6000
5700
5700
6200
CASTLE ROCK
SOUTH TABLE MOUNTAIN
CLEAR CREEK CANYON
6000
N
99
1000 METERS
AIR SAMPLE LOCATIONS
A PERMANENT CPC
PERMANENT RFP
HIVOL SAMPLES INTERMITTENTLY
PLANT ELEVATION 5740 - CONTOUR ELEVATIONS IN FEET

as determined on a monthly basis, exceeds 0.01 micrograms per cubic meter, but does not exceed 0.05 ug/M^3, the plant will inform the AEC of specific procedures which will be undertaken to reduce the airborne concentration. In the event that concentration exceeds 0.05 ug/M^3, operations will be immediately halted and the necessary corrections made to reduce the average concentration to below 0.01 ug/M^3. In any event, concentrations above 0.01 ug/M^3 will be permitted to exist for not more than a 60 day period unless specifically authorized by the Commission. Such authorization will be forthcoming only if steps are bei... taken which are expected to result in a satisfactory ... in effluent material.

In the event that the maximum average neighborhood concentration of uranium at the ground during any calender month, as determined on a monthly basis, exceeds 4 x 10^{-12} uc/cm^3, or 4 x [illegible] uc/cm^3 when thorium is being processed, the plant will inform the AEC of specific procedures which will be taken to reduce the airborne contamination. Further actions will be at the direction of the AEC.

[illegible] PLANT WATER SAMPLING

[illegible] four [illegible]inage systems which might be affected by our operations.

Sucker Gulch. This is a small stream which flows out of Golden Gate Canyon to the [illegible]est of the plant site. It is joined at a distance of approximately 1200 meters north west of the plant site by -

Cressman's Gulch, a small intermittent stream. After confluence, the stream flows within 50 meters of the plant site and drains into -

Clear Creek. This is a large stream flowing from Clear Creek Canyon through Golden in a northeasterly direction.

Church Ditch. This is an irrigation canal which originates at Clear Creek west of Golden and flows open to the edge of the plant site. It then goes underground around the edge of the plant site and continues around North Tabl Mountain to the northwest.

Water samples will be collected at points 2000 and 1000 meters above and below the plant site and at the plant site. A sample point will be established at the confluence of Tucker Gulch and Clear Creek.

These points will be sampled weekly for four weeks to establish background levels before plant start-up. After operations commence, samples will be collected from each of the established sampling points monthly on a schedule which requires each [illegible] system to be sampled weekly unless in-plant conditions dictate otherwise.

10 Feb 61

All water samples will be analyzed spectrographically for beryllium. A tributylphosphate extraction will be performed and the samples then counted for alpha activity in a scintillation alpha counter. The geographic location of the sample locations is shown on the accompanying diagram.

VI. CONTAMINATION CONTROL

A. Protective Clothing

Protective clothing will be provided for all personnel actively processing beryllium and/or uranium. The normal protective clothing for all production personnel working in the processing and grinding phases will be outer garment, (coveralls or dress), socks, and safety shoes. The garments and socks will be changed daily. Personnel working in the Inspection, Laboratories and services will be provided smocks or laboratory coats, socks, and safety shoes. The smocks will be changed on a weekly basis, the socks daily. Visitors to the area will be provided smocks and shoe covers.

Separate facilities are provided in the clothing change area for street clothing and work clothing. These facilities are physically separated to prevent the possible spread of contamination to street clothing from the work clothing. Used protective clothing will not be allowed beyond the portion of the locker room provided for changing protective clothing.

Used protective clothing will be shipped to a commercial laundry service equipped and approved for laundering contaminated clothing.

B. Personnel Decontamination

Showers will be required of certain personnel working in the processing and grinding operations on a daily basis. Personnel in the Laboratories, Inspection and Services will be provided showers on an optional basis.

Washing of the hands and face is the minimum clean up allowed before lunch and breaks.

Personnel decontamination will be carried out under the direction of the Health Physicist or the Health Physics Technician in the area. Decontamination agents such as detergents, scrub brushes and chemical agents will be located at the decontamination sink.

A hooded enclosure and decontamination sink is provided for material and equipment decontamination. Chemical and physical decontamination supplies are available at this location. Health Physics will determine amounts and locations of contamination and monitor decontamination operations.

C. Contamination Surveys

Routine smear surveys are made in the process and administrative areas to determine uranium and beryllium contamination.

Whenever contamination levels are found which exceed the working limits, decontamination will be effected immediately.

Area decontamination will be accomplished by vacuuming, mopping, and/or wet scrubbing with a power scrubber, followed with a pickup of the water by an industrial vacuum cleaner equipped with an absolute type filter on the air discharge.

Surveys will be made on all outgoing materials and equipment to assure that no contamination leaves the process area.

Periodic alpha surveys of equipment and areas are made. Wherever enriched uranium is in the process these surveys will indicate equipment leakage and failure.

D. Maximum Permissible Working Limits

1. Alpha Activity

The maximum levels for alpha contamination on materials leaving the control of the process area or in the administrative areas of the building are:

50 $d/m/ft^2$ removable by smearing.

250 c/m direct as measured at 2 Pi geometry.

The maximum levels for alpha contamination within the process area are:

250 $d/m/ft^2$ removable by smearing

500 c/m direct as measured at 2 Pi geometry.

2. Beryllium Contamination

The maximum permissible levels for surface contamination removable by smearing are:

Floors and walls	0.02 ug/cm^2
Unenclosed equipment	0.01 ug/cm^2

3. The following values are being used for the Maximum Permissible Concentrations for airborne contamination:

Beryllium:	[illegible] micrograms per cubic meter for continuous exposure with a maximum concentration of 25 micrograms per cubic meter for any period of time no matter how short. These figures are based on the Atomic Energy Commission recommendations, and the Hygienic Guide series of the American Industrial Hygiene Association.
Uranium:	10^{-10} microcuries per cubic centimeter.
Thorium:	[illegible] microcuries per cubic centimeter.

E. Analytical Methods

All alpha determinations on air samples and smear samples will be made in a scintillation type alpha counter. This instrument is the Eberline Instrument Corporation Model SAC-2. This instrument operates at a background level of approximately 0.1 c/m with a geometry of 40%.

General air samples will be counted once. If more than the MPC_a is indicated on the sample, the sample will be counted a second time a minimum of four hours later. Koval's formula is then applied to the two counts to give the long lived activity after correction for short lived activ[illegible] from the Radon-Thoron decay products.

Special air samples and duct samples will be counted twice and the Koval determination for long-lived activity applied.

Beryllium analysis of air samples and smear samples will be performed using an emission spectrograph. Sample preparation procedures will involve the Silver [illegible] Carrier Method develop[illegible] by the Dow Chemical Company for the determination of beryllium in air filter samples.

F. Contaminated Waste Disposal

Liquid [illegible] will be collected in four drainage systems. Criticality considerations of these systems are discussed elsewhere.

All liquid drains which originate in areas where there is no possibility of contamination drain directly to the domestic sewer. These drains originate in the administrative and office areas.

Liquid drains which originate in the process area where there is a possibility but a low probability of contamination, will be collected in 1000 gallon holding tanks where it can be sampled. If the samples indicate levels below 1.0 ppm for beryllium and 3×10^{-5} uc/cm^3 for uranium, or uranium-thorium mixtures, the liquid will be released to the domestic sewer system. If the samples indicate levels above this, the liquid will be decontaminated in our low level waste treatment facility by flocculation, decantation and filtration. The decontaminated liquid will again be sampled to assure levels below the above stated levels before being released to the domestic sewer system.

Liquid waste which originates in areas such as the decon[illegible] sinks, filtrate drains, etc., where there is a high probab[illegible] of contamination above the allowable levels for sewer discharge, will be collected in 1000 gallon holding tanks. Liquid in these tanks will be decontaminated in our own low level liquid waste treatment facility. Decontaminated liquids will be sampled to assure contamination levels below those indicated above before being released to the domestic sewer.

Liquids which originate where enriched uranium solutions are processed will be collected in critically safe vessels for ultimate recove[illegible]

Solid waste originating in the process area which contains non-recoverable amounts of beryllium, uranium and/or thorium will be collected in closed containers lined with plastic bags. This will be primarily such materials as paper, wiping rags, etc. The bags, after being sealed, will be packed in metal drums for ultimate disposal in accordance with instructions from the AEC.

Solid waste containing recoverable amounts of enriched uranium will be collected in covered, critically safe containers for shipment to recovery processing areas.

C. Smoking and Eating

No smoking or eating will be allowed in any area where radioactive and/or beryllium compounds are processed or handled in any form. Specially designated areas are set aside for smoking areas. The portion of the locker rooms where protective clothing is allowed is designated as a smoking area. Washing facilities are available in this area and all personnel are [illegible] to wash their hands prior to smoking in this area.

VII. RADIATION DOSIMETRY

A. Film Badges

The greatest radiation hazard from enriched uranium will arise in the event of an accidental nuclear excursion. It is impossible to predict what personnel might be involved in such an incident. Therefore, all personnel in the process area are required to wear film badges containing both low level and high level beta-gamma sensitive film.

Similar film badges will be provided for all visiting personnel not normally assigned to the area.

A commercial film badge supplier will furnish the film badges and processing. This firm will also keep a set of permanent records on all exposures. This is in addition to the records maintained by the Health Physics group. The sensitivity range of the film furnished is 10 mrem to 550,000 mrem.

Complete compliance with the AEC requirements will be assured by shutting down any operation and removing the operator from further exposure if any individual receives more than:

3 Rem penetrating exposure in a 13 week period, or
6 Rem skin dose in a 13 week period, or
15 Rem penetrating dose in a year's period, or
30 Rem skin dose in a year's period, or
An accumulation which exceeds an average of 5 Rem penetrating dose for each year of age past the age of 18.

It is not expected that any operation performed in conjunction with this project will yield radiation exposure which will even approach these levels. Experience on similar operations at other installations has shown very little radiation problem.

B. Gamma Alarms and Threshold Detectors

A gamma alarm system is provided with five detecting stations. Each of these stations is located optimally with respect to detecting radiation from a criticality. The processing area will have two detectors, the inspection area one, the furnace area one, and the storage vault one.

A central follow meter panel with individual readout meters for each detector is coupled with a master alarm panel and follow meter. In the event that the radiation at any detector reaches the level set on the alarm panel, an alarm light and distinctive audible alarm will be indicated at the detector. This will be the signal to personnel in the area to evacuate as rapidly as possible. At the same time, an alarm will sound at the follow meter panel. This panel is located in the foyer near the main entrance. This location allows reentry personnel to determine radiation levels in the process area without having to enter that area.

The alarm levels set at each detector will be 15 mr/hr.

Special film badges with beta-gamma film, NTA neutron film and threshold detectors of indium, sulfur, gold, and cadmium covered gold, will be placed around the process area. These area monitoring film badges will provide information about radiation intensities and spectra.

VIII. MEDICAL PROGRAM

Physical Examinations

A pre-employment physical examination is required of all employees working on this project. This places special emphasis on previous work history, complete blood and urine analysis, chest X-ray, vital capacity, and history of lung, kidney, or bone troubles.

Monthly measurements of weight and vital capacity will be made.

A semi-annual physical examination with a chest X-ray will be required of all personnel actively processing material. Annual physicals will be required of supervisory and administrative and maintenance personnel. A complete physical will be required upon termination by an employee.

[illegible] [illegible]nts of upper respiratory irritation will be immediately [illegible] to a physician for examination.

[illegible] the process area will be reported and [illegible] examination. A trained First Aid [illegible] for [illegible] wounds.

A physician with a security clearance is available on call to treat injured persons. Open wounds will not be allowed in the process area.

B. Urinalysis

An initial urine sample will be requested of all employees. Periodic urine samples will be collected from all personnel associated with this project. These samples are submitted to a commercial bioassay laboratory for uranium analysis by the ether extraction alpha counting method.

Special samples may be required when exposure of an individual is suspected.

IX. PROCESS STATIONS

Process stations are discussed in detail on pages 22 through 33 in the classified copies of this manual.

23 thru 32

These pages are included in CPC-69. This document is unclassified without the inclusion of these pages.

3 Apr 61

X. RECORDS

A. Operating Personnel Exposures

A complete set of records will be maintained covering all phases of potential exposure to the workers within the plant. Records will cover:

1. Routine air contamination levels within the plant.
2. Air contamination levels as a result of special operations, spills, etc.
3. Contaminated accidents or incidents which might cause personnel exposure.

4. Urinalysis results.

5. Film Badge readings.

6. Periodic Physical examinations.

B. Environmental Exposures

Records will be kept on all evaluations made which might have some effect on the plant envorons and nearby residents.

1. Off-site air samples.

2. Samples of effluent plant air.

3. Samples of effluent plant wastes.

4. Water and soil samples off-site.

XI. MATERIAL SHIPPING AND PACKING

All process materials to be shipped will be packaged to conform to AEC and ICC regulations governing shipping of radioactive materials. Shipping containers will be checked for contamination before shipping. Decontamination facilities are available to assure complete removal of contamination.

XII. PERSONNEL TRAINING PROGRAM

All employees entering the project will be required to participate in a training program related to Health and Safety. The meetings will be of the conference type, with direct employee participation. The initial phase of the program will be a meeting to present all of the phases of the Health and Safety program as outlined below. This initial contact will be followed up by further periodic meetings to review the [illegible] phases individually and to test the employees on their knowledge of the program. Periodic practice evacuations will be held to assure that employees recognize alarms and know evacuation routes and procedures. After successful completion of the training program, which will take approximately 8 hours per employee, each participating employee will be given written recognition of his participation and qualification.

The following outline will be used as a guide in preparing the training program. The Health Physicist for the project will conduct the program.

I. Internal problem of toxic and radioactive materials.

Methods of entry to the body are by:

A. Inhalation. Preventative measures include:

1. Confinement

a. Glove Box

b. Velocity Hood

B. Ingestion

1. No smoking rule.

2. No food or edibles in process area.

C. Injuries

1. Safe practices prevent injuries.

2. All [illegible] must be reported.

II. External problem of radioactive material.

A. Film Badges

1. Description of method

2. Reason for need

B. Criticality Alarms

1. Purpose and function

2. Location of alarms

3. Visual and audible signals.

4. Evacuation procedures and routes.

III. Medical Program

A. Pre-employment physical examination

B. Periodic re-examination

C. Urinalysis program

IV. Alpha Activity

A. Characteristics

B. Method of detection

V. Basis for MPC levels

A. Maximum level for continuous exposure without harm

B. Exceeding MPC for short periods is undesirable, but not immediately dangerous.

VI. Safety Program

A. Company policies

B. Compensation laws

C. Importance of reporting all injuries

10 Feb

VII. Definition of responsibility

A. Management's responsibility to provide safe working conditions.

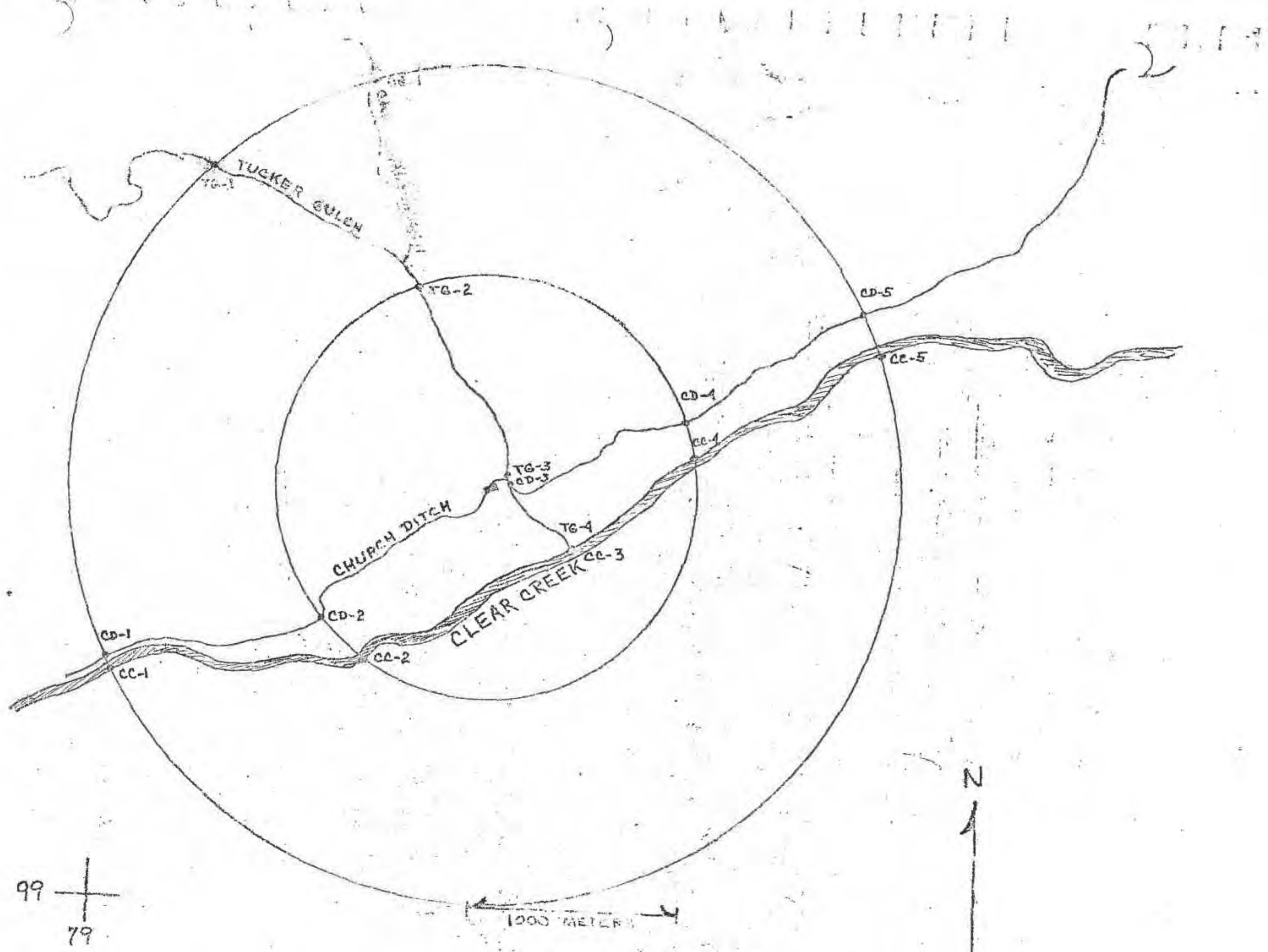
B. Health Phys[illegible] responsibility to assure that conditions are safe.

C. Individual's responsibility for compliance with rules and procedures.

XIII. EMERGENCY COVERAGE

A fully equipped and trained monitoring team operating under the Radiological Assistance Plan is available to provide emergency coverage in case of major emergencies. This team is located at the Rocky Flats Plant of the USAEC.

[illegible]



WATER SAMPLE LOCATIONS

APPLICATION FOR AEC LICENSE TO
RECEIVE, STORE, PROCESS, AND SHIP
SPECIAL NUCLEAR MATERIAL

APPENDIX B

NUCLEAR SAFETY

January 20, 1964

Coors Porcelain Company
Golden, Colorado

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CONTENTS

APPENDIX B - GENERAL

General -

This portion of the license is for the purpose of handling, processing, and storing uranium (enriched in U-235) compounds and solutions. Uranium metal, UF_6, uranium (enriched in U-233) compounds, and Plutonium metal and compounds are not considered in this license application.

Nuclear safety is concerned with the protection of Coors Porcelain Company personnel, plant facilities, and the surrounding communities from the hazards of radiation and contamination associated with a criticality accident. Maximum effort is directed toward the prevention of criticality accidents, however, an accident is possible, and the protection includes a plan for handling this type of accident.

Nuclear safety is the responsibility of the Division Manager. The Division Manager in turn vests each of his supervisors with the nuclear safety responsibility for the actions of his subordinate supervisors and reporting personnel. The Nuclear Safety group, which is a part of the Health and Safety Group, is an advisory staff to the Manager and his line supervisors. The Nuclear Safety group consists of a Nuclear Physicist and Nuclear Safety Inspectors. The inspectors monitor U-235 handling, processing, and storage areas to assure nuclear safety. The Nuclear Physicist directs the nuclear safety program and has the authority to stop any operation which may create a nuclear safety hazard.

The basic operating principle of nuclear safety is that an isolated unit of special nuclear material containing not more than 350g of U-235 of 95% enrichment is safe under any condition which might exist in a development or production facility. Under well controlled conditions this basic unit of material can be increased. Factors which must be considered when the mass of the unit is increased are: U-235 enrichment, size, shape, density, dilution, moderation, reflection, and nuclear poisons.

Administrative control over the unit will be in effect at all times. Administrative control will be supplemented by geometry control and/or moderation control whenever the processing, handling, or storage of U-235 lends itself to this type of control.

If criteria for nuclear safety other than that in proposed 10 CFR Part 70 is used, a safety factor greater than two will generally be used. New criteria, if used, will come from critical mass experiments. Critical mass calculations will be used only as back-up criteria, not as fundamental criteria.

The facility contains a 29-foot by 160-foot production area, a 60-foot by 160-foot production and service area on the ground floor, and a 60-foot by

160-foot second floor for the laboratory and service areas. At the back end of the building is a 20-foot by 89-foot depressed area. A tunnel runs under the floor of the production area into the depressed area.

The tunnel and depressed area are below grade and could be flooded. The likelihood of flooding from external sources of water is very low as the depressed area walls project a minimum of one foot above grade. The rest of the facility is above grade where flooding is not credible. Any water in the building above grade would flow into the depressed area or out of the building.

All process low level waste ($<$20 ug u/g liquid) liquids are collected in holding tanks in the depressed area. The waste liquids are disposed according to Health Physics procedures in Appendix A.

All solutions containing recoverable amounts of U-235 ($>$20 ug u/g liquid) are collected in 4-inch diameter bottles in the tunnel. The bottles are spaced more than twelve inches apart in a line array to allow for flooding.

Grinding sludge containing U-235 is filtered and stored in the depressed area. Unit batches of U-235 sludge are stored with 12-inches minimum separation to allow for flooding.

This is the extent of processing below grade where flooding is possible.

10 CFR 70.25 - Nuclear Safety

(a.1) The possibility of criticality exists at most process steps with the exception of certain pieces of equipment described in Part I of this appendix. However, the probability of criticality is quite low because of the combination of several factors. These factors include the safety factors in the nuclear safety criteria of The Nuclear Safety Guide, TID-7016, Rev.1, and proposed 10 CFR 70, the process control resulting from close supervision, and the continual monitoring by members of the Health and Safety Group.

(a.2) The equipment not described in Parts I, II, and III of Appendix B depend on unit mass control to assure nuclear safety at each process step. The equipment described in Parts I, II, and III of Appendix B depend primarily on geometry control for nuclear safety. Other equipment in Parts I, II, and III (storage racks, shipping containers, etc.) depend on a combination of unit mass and geometry control.

Because the U-235 isotopic enrichment can vary for different production contracts, the unit mass and geometry controls have been established for three enrichment ranges. Table I lists the controls and nuclear safety limits for the three enrichment ranges for a U-235 compound or solution.

Table I

Control	0.70 to 5% U-235	5.01 to 20% U-235	20.01 to 95% U-235
U-235 Unit Mass, g	770	420	350
Diameter of Infinite Cylinder, in.	10	6	5
Thickness of Infinite Slab, in.	4.8	2.4	1.5
Volume, liters	25.4	10.2	4.8

Hereafter, a unit will refer to the discrete aggregation of U-235 allowed in a container or piece of equipment which depends on mass control for nuclear safety. This quantity as listed in Table I has three different values for the three isotopic enrichment ranges.

The controls listed in Table I can be adjusted upward when additional controls of degree of moderation, fixed neutron poisons, minimal reflection, or effective density can be applied.

When beryllium compounds are used, the unit mass will be limited to less than 300g of U-235. The other controls in Table I will remain unchanged. Heavy water and graphite are not considered in this application.

Administrative control which consists of established and approved process procedures and close supervision will be in effect at all times to prevent the breakdown of any nuclear safety control.

(a.3) An accidental condition of criticality is highly unlikely because:

(i) Generally the simultaneous failure of two independent safeguards is required for an unsafe uranium configuration. The employment of safety factors mentioned in (a.1) requires more than one safeguard failure for criticality.

(ii) Administrative Control (see a.2) includes the efforts of Management, Supervision, and members of the Health and Safety Group. Nuclear Safety Inspectors from the Health and Safety Group monitor all areas where special nuclear materials are processed, used or handled to assure nuclear safety.

(b.1) Procedures for receiving U-235 compounds include the following:

(i) After the special nuclear material is received from the carrier, individual packages will be taken to the sampling area to obtain a sample for Laboratory analysis of the degree of moderation, chemical composition and isotopic content.

(ii) The unit package will then be weighed and stored until the analysis results are returned from the laboratory. When the analysis is obtained, the net weight of special nuclear material in each package can be calculated. The material can then be transferred to storage containers and moved into regular storage areas or into the process area.

(iii) The packages as received from the carrier in the receiving area and in the sampling area will be stored in an array which has safety equal to or safety greater than the safety of the array on the carrier vehicle. The array or arrays will be isolated from other arrays in accordance with 10 CFR 70 (par. 57).

The handling and storage of these packages till be done in accordance with written procedures approved by Nuclear Safety and will be monitored by member(s) of the Accountability Group and member(s) of the Health and Safety Group.

Storage limits and procedures will be in accordance with the next section.

(iv) Wet and damaged packages of U-235 compounds will be received in the following manner:

Wet packages will be handled in the receiving and sampling areas in the same manner as described in (iii) above.

Damaged packages which have not suffered loss of contents will be handled in the manner described in (iii) above. Damaged packages which have suffered partial loss of contents will be repackaged in volume safe containers and be held for instructions from Accountability.

Either type, or a combination of wet and damaged package, will remain with the array being received, but will be handled and placed so as to keep equal or greater safety in the array as on the carrier.

10 CFR 70.25 - Nuclear Safety

(b.2) Procedures for the storage of U-235.

(i) The total amount of material at Coors Porcelain Company will not exceed 1000 kg of U-235 at any one time. The actual amount of material in process will be much less than this; probably less than 100 kg of U-235 at any one time.

A series of different sized containers will be used for different U-235 enrichments and for different amounts of diluents in a unit. The containers will be color coded to handle only a certain type of material. Administrative control will be used to assure proper container selection. Volume control will be effected by proper selection of a unit container.

A minimum edge to edge separation of 12-inches will be maintained at all times between units. The interaction between units will be limited to comply with the solid angle criteria in 10 CFR Part 70.52.

(ii) The containers which are currently used are:

<u>10# Fruit Can</u>

This is a 7-inch diameter by 9-inch high metal can with a lid which is <u>not water tight</u>. It is limited in use to a basic unit of U-235 plus moderator or diluents with increases in unit mass allowed only for decreases in isotopic enrichment.

<u>30# Fruit Can</u>

This is a 10-inch diameter by 12-9/16-inch high metal can with a lid which is <u>not water tight</u>. It is limited in use to a basic unit of U-235 plus moderator or diluents with increases in unit mass allowed only for decreases in isotopic enrichment.

<u>4-inch I.D. by 40-inch Long Cylinder</u>

This cylinder is safe for any U-235 solution concentration.

(iii) For a description of the storage facilities see Appendix B, Part 1.

(iv) Flooding of storage areas is not considered credible. However, the minimum spacing between units is 12-inches so that flooding would not create a more reactive situation. Unless water tight containers are used, the most effective degree of moderation is assumed for units in storage.

(v) Units of U-235 will be taken to or from the storage vault only by vault personnel. The unit going into storage must be accompanied by a laboratory analysis sufficient to determine the U-235 weight. If the mass limit of the unit going into storage has been increased because of moderation control, the material must be analyzed for moderator and diluent content in addition to the U-235 weight analysis.

(b.3) The general equipment layout and storage areas are shown in Coors Porcelain Company Drawing Nos. 44-0400-7 and 44-0400-8 in Appendix A. The general process is described in the General section of this license application. (See 70.23 para 3, page 21.)

With few exceptions, the process is a batch type operation with the U-235 unit mass limit a controlling factor. Some mass limited process steps or process area need some additional explanation. These are:

(a) U-235 solutions are introduced into a ceramic body matrix through a fuel tower designed to prevent "double-batching". The U-235 is measured in a safe volume container on a balance and then drained into the ceramic body mixer. However, the fuel balance cannot be operated unless the fuel balance drain valve is closed, the mixer is empty and the mixer drain valve is closed. The fuel balance can be filled once for each mixing operation. This prevents dumping two units of U-235 into a mass limited mixer. This fuel tower is controlled by electrically operated valves which can be activated only in the proper sequence controlled by memory devices. This control system has been working for more than eighteen months.

(b) The three areas of the laboratory-wet chemistry, x-ray emission spectrograph, and arc-spark emission spectrograph areas and the development area in the same room with wet chemistry are each limited to a unit equivalent. These areas will be handling samples from many units. Each sample will be considered a fraction of a unit. The sum of the fractions will not exceed unity in each area.

(i) The description of equipment in which the hazards of criticality exist because more than one unit is being processed at one time are described in Parts I, II, and III. The more credible hazards and hazard controls are described in Part I, Appendix B.

(ii) Because this application is for the general purpose of handling U-235 for development work as well as production work, the only limitation on the U-235 is that the material will be a compound or solution, not in a metallic or gaseous state.

(iii) The maximum quantity of U-235 at any one time will be limited to one unit of material at a piece of processing equipment with the following exceptions:

Two units can be at a process table when the units are separated by a physical barrier at least 12-inches thick.

More than one unit can be at a piece of processing equipment employing a combination of geometry and administrative control. (See Part I for a description of this equipment.)

The total amount of U-235 at any step of the process is a function of the number of pieces of equipment under unit control or the size of the equipment under geometry control.

(iv) Spacing of unit masses of U-235 within each process area will be not less than 12-inches edge to edge except for equipment noted in Part I. The normal spacing is usually much more than 12-inches due to the spacing for aisles, etc., between different pieces of process equipment.

Separation between adjacent process areas is usually limited only be an aisle or a cinder block wall. Bays in the vault are separated by 8-inch thick high density concrete walls.

(v) Material from one process operation is placed in a container for transportation to the next process operation or to in-process storage until the next process station is ready to receive another unit of material. The material in a container is placed in a transport cart and wheeled to its proper destination. The cart has been designed to keep a minimum distance of 12-inches edge to edge separation between units on carts or between a unit on a cart and units in a storage rack. When a unit of material is carried up or down stairs, a portable birdcage is used. There are a few places where a portion of the unit is carried up or down stairs without a spacing device such as a cart. An example of this type action is the hand carrying of unit samples to the laboratory.

Administrative control will be used to assure that only one enrichment will be in a process area at a time.

(vi) Gamma surveys are made at least weekly on containers which are supposed to be free of U-235, except for contaminating amounts on plastic bags, paper, etc.

The grinding sludge is controlled by use of safe geometry tanks. In addition, the number of elements ground are tabulated to limit the amount of sludge material accumulating in a safe tank.

(b.4) Procedures which can be utilized for material control to assure compliance with provisions in this license include:

(i) Percent Solids. This determination can be done in the Laboratory or in the Ceramic Body Preparation Area. This type analysis yields ceramic body unit weight, and moderator and other combustible material weight mixed with the ceramic body. This determination is made by weighing, heating and re-weighing on laboratory-type equipment. This analysis is normally done on "received" material, on units being mixed with binder, etc., and on scrap which is being recycled for reprocessing.

(ii) Isotopic Enrichment and Content. This determination is done in the Laboratory. An Arc-Spark Emission Spectrograph yields two significant figures for isotopic enrichment analysis results which is adequate for nuclear safety. This type analysis will normally be done only on "received" material.

Isotopic content is determined by applying isotopic enrichment factor to the total uranium content which is determined by wet chemistry methods or by X-Ray Emission Spectroscopy. This analysis is done on "received" material, and on units during ceramic body preparation.

(iii) Assay of Waste. Waste is defined as non-special nuclear material contaminated with not more than 20 ug of U-235 per gram of non-special nuclear material. All waste containers in the process areas are surveyed for radiation at least weekly. All waste containers also are surveyed for radiation before leaving the building for burial shipment.

Liquid wastes are analyzed by means of a gamma scintillation spectrometer to determine correct means of disposal. See Health Physics requirements in Appendix A for disposal procedures.

(b.5) Monitor Alarm System and Emergency Procedures

The radiation monitor alarm system and emergency procedures are described in accordance with the standards set forth in 10 CFR 70.34, Monitor Alarm System and 10 CFR 70.35, Emergency Procedures.

10 CFR 70.34 - Monitor Alarm System

(a.1) The monitoring system has a preset alarm level of 10 millirems per hour.

(a.2) The monitoring system has a response time less than 3 seconds at a radiation level of 20 milliroentgens per hour. The time constant for the alarm circuit is 1.8 seconds.

(a.3) The monitoring system is capable of operating the alarm when the radiation level at a distance of 1 foot from the location where special nuclear material is handled, used or stored, which is most distant from the sensing device exceeds 300 roentgens per hour.

(a.4) The monitoring system is capable of operating the alarm at the radiation level anticipated from an incident causing 10^{18} fissions during a period of 0.1 second at the point where special nuclear material will be located nearest to the sensing device.

(a.5) A sensing device is within 120 feet of every location where special nuclear material is handled, used or stored, or at such lesser distance as may be required to compensate for intervening shielding. See Coors Porcelain Company Drawing Nos. 44-0400-3 and 44-0400-5 for location of the sensing devices. Calculations were made of the gamma intensity at the detector by the following method:

$$I = \frac{(I_1 D_1^{\,2})}{D_2^{\,2}} e^{-ux}$$

Where $I_1 =$ 300 rem/hr

$D_1 =$ 1 foot

$D_2 =$ Distance to detector in feet

$ux = (u_1 x_1 + u_2 x_2 + u_3 x_3)$

For 700 Kev gamma the following data was used:

	u
Air	0.00009
Cinder block	0.142
Steel	0.548
Concrete	0.176

A sample calculation within the vault at station (1) is illustrated below, based on the above criteria. Assuming a radiation intensity of 300 rem/hr at one foot at point (1), attenuation of radiation to the detector would be imposed by air, high density concrete and steel.

10 CFR 70.34 - Monitor Alarm System

Air	30' = 918 cm
Concrete	8" = 20.4 cm
Steel	1/8" = 0.32 cm

$$ux = (0.00009 \times 918) + (0.176 \times 20.4) + (0.32 \times 0.545)$$

$$ux = 3.87$$

$$I = \frac{300}{900} e^{-3.87} = 15 \text{ mr/hr}$$

This 15 mr/hr would be sufficient for detection by a probe monitoring radiation above the 10 mr/hr alarm level set on the system.

The results of 24 locations chosen on the basis of maximum credible distances and shielding conditions are tabulated below, accompanied by their air distances and assumed shielding. The locations of these points with respect to the detectors are illustrated on the accompanying drawings - 44-0400-3, and -5.

(1)	15 mr/hr	30' of air, 8" concrete, 1/8" steel
(2)	385 mr/hr	27' of air
(3)	385	27' of air
(4)	71	31' of air, 1" steel

(5)	19 mr/hr	22' of air, 12" concrete
(6)	16	27' of air, 8" brick
(7)	12	71" of air, 1" steel
(8)	91	14' of air, 2" steel
(9)	91	14' of air, 2" steel
(10)	265	7' of air, 1" steel, 6" concrete
(11)	104	50' of air
(12)	53	58' of air, 1/4" steel
(13)	55	67' of air
(14)	21	33' of air, 3/8" steel, 6" concrete
(15)	21	30' of air, 3/8" steel, 6" concrete
(16)	26	27' of air, 1/8" steel, 6" concrete
(17)	258	30' of air, 1/8" steel
(18)	503	20' of air, 1/4" steel
(19)	202	38' of air
(20)	28	67' of air, 1/2" steel
(21)	57	31' of air, 6" concrete, 1" steel
(22)	86	40' of air, 1/2" steel
(23)	43	56' of air, 1/2" brick
(24)	62	56' of air, 6" concrete

Coors Porcelain Company
Drawing Number 44-0400-3

Coors Porcelain Company
Drawing Number 44-0400-5

10 CFR 70.34 - Monitor Alarm System

(b.1) Each monitor alarm circuit is equipped with an auxiliary power source which will automatically supply the system in the event of disruption of primary power.

(b.2) Each monitor alarm circuit is tested by sounding the alarm at the time of the practice evacuation drill.

(b.3) Each monitor-alarm circuit has a red light alarm on the control unit in the event of low voltage at the sensing device or low B^+ on the control unit. Each monitor-alarm circuit with the exception of the audible alarm will be tested weekly with a built-in Sr^{90} check source located in each monitor.

(c) The alarm is clearly audible in all portions of areas in which special nuclear materials are handled, used or stored.

(d) The alarm system is designed and constructed so that the alarm will continue to sound until re-set by the designated supervisor.

10 CFR 70.35 - Emergency Procedures

Evacuation drills are in accordance with standards set forth in 10 CFR Part 70.35, Emergency Procedures.

The training of all employees includes a written set of instructions to follow in the event that the radiation alarm is sounded. Additional training includes practice evacuation drills for personnel on all shifts. These drills, which will be conducted at least once every three months, are conducted in conjunction with tests on the audible radiation alarm.

When the audible radiation alarm sounds, all personnel are to proceed to the nearest exit for the area in which they are working without stopping to change clothing, shoes, or film badges. All personnel are to proceed to the designated assembly area and report to the area emergency warden.

The area warden will then determine that all the people are accounted for. The area warden will also attempt to determine the location and cause of the radiation alarm from the people in his area. At this point the area warden will report his findings to the chief warden who in turn reports to Management. On normal operating shifts, the chief warden is a member of Management or top supervision.

10 CFR 70.35 - Emergency Procedures

When the radiation alarm sounds, the security guard is to move the emergency re-entry equipment from the guard lobby to the outside of the building. The guard is to maintain security surveillance and prevent unauthorized personnel entering the building.

The Chief Warden, the Health Physicist, and Management will evaluate the situation and take necessary measures for re-entry into the building.

10 CFR 70.26 - Combined Operations

All contract and/or licensed material will be handled and stored under provisions of this license with the exception of contract material in contract-furnished shipping containers. Contract material in these containers will be isolated from licensed material according to the criteria outlined in 10 CFR 70.57. The containers for receiving and shipping contract material are exempted from the provisions of this license application.

APPENDIX B, PART I

A brief description of the equipment not limited to a basic unit of mass by reason of moderation and/or geometry control follows:

1. Oven (Exhibit III-1, Appendix B, Part III)

   The oven's purpose is to dry batches of wet ceramic material (filter cake). The filter cake is placed in 24-inch diameter by $1\frac{1}{2}$-inch high pans. A loose fitting lid covers each pan. Sixteen pans are placed in the oven for a drying cycle. The pans are placed in a 4 x 4 vertical array on 16-inch vertical centers. The pans have 16-inch horizontal edge to edge separation for the two middle pans on each shelf. However, there is no horizontal separation between the outside and middle pans on each shelf.

   Geometry control is used here to control the spacing of the pans; fixed barriers assure the separations mentioned above. Administrative control is used to assure that not more than one unit of mass is in each pan. Moderation control is not effective at this point because moderator content is not controlled by the prior filtration step.

   The oven is cooled by moving air. Flooding is not credible; however, the array would be safe under flooded conditions because of the vertical separation between pans. Water hold-up after a flood is no problem because of the height of the pan.

Over two thousand units were cycled through this unit in a six month period during production of Tory II-C fuel elements. No procedural or administrative control breakdown was detected.

2. Scrap Recovery System (Exhibit III-2, Appendix B, Part III)

Some of the raw materials in process ends up as recoverable scrap material. This material, after recovery, re-enters the process stream in the unit weigh-out area.

The recovery process consists of collecting the scrap material in recovery units. The unit size is carefully controlled through chemical analysis, weight, and administrative control. The unit of material is then pulverized, slurried, and milled. After milling, the slurry is stored in a system of five 4-inch I.D. by 20 foot pipes spaced on 2-foot centers. The slurry is stored in the pipes until the material re-enters the process stream.

Geometry control is used to assure nuclear safety of the slurry in the pipes. Administrative and unit control are used to assure nuclear safety of the scrap material before the material is slurried and stored.

Flooding of the system is not credible; however, the recovery system is safe against flooding.

This system is in operation and assures nuclear safety mainly by geometry control.

3. Furnace - Hevi Duty (Exhibit III-3, Appendix B, Part III)

This furnace is used to burn binder out of ceramic material or parts and to dry grinding sludge. The furnace is an open air type which is an electrically heated and water cooled.

Parts are placed on saggers and enter the heating zone on a moving chain link belt. The cross-sectional opening is $6\frac{1}{2}$-inches high by 13-inches wide. The saggers are butted together to form an essentially infinitely long slab. Trays of sludge or ceramic material containing a unit or less of special nuclear material are also run through the furnace when parts are not undergoing binder burn-out. A physical barrier one foot or more in length separates trays of one unit from trays of a second unit.

Geometry control is used to control the height of material on the saggers or in the trays. Geometry control is effected by the design of the sagger, the height of a tray, and/or a maximum profile mask over the opening to the heating zone.

Water flooding is not credible because the furnace is an open air furnace.

If material is not removed before it reaches the end of the belt at the exit port, a micro-switch is thrown by material on the belt, which stops the belt. This prevents stack-up of material at the end of the furnace.

This furnace has operated successfully and safety during a six month period of full production of Tory II-C fuel elements.

Furnace - Harper (Exhibits III-4, III-5, Appendix B, Part III)

The furnace is used to fire ceramic parts in a hydrogen atmosphere. The furnace is electrically heated and is cooled by means of circulating water. The firing chamber is 6-inches high by $7\frac{1}{4}$-inches wide. Saggers are butted together to form an infinitely long array. The height and amount of ceramic material in each sagger is limited by administrative and procedural controls.

Nuclear safety is effected by geometry control and moderator control. All binder material and hydrogenous moderator is removed in the Hevi Duty furnace so that moderators are very highly controlled at this point.

Flooding of the firing chamber is not considered credible. Because of the explosion hazards involved with high temperatures and a hydrogen atmosphere, the cooling water pressure and chemical activity is closely controlled. This is done to prevent a pressure rupture or chemical corrosion to the cooling jacket. The drain on the furnace is open so that pressure cannot build up and rupture the cooling jacket.

The furnace is checked weekly for hydrogen leaks to prevent any accumulation of hydrogen in the building. It is also checked after the furnace has been rebuilt (usually every 9 operating months). Operating experience during the past two years has shown this procedure to be adequate for hazards control.

4. Storage Racks (Exhibits III-6, III-7, III-8, III-9, Appendix B, Part III)

Unit containers are stored in "pigeonholes". An array of "pigeonholes" constitutes a storage rack. The storage racks are constructed from metal lumber for support; from expanded metal or rods for unit separators; and sheet metal for "pigeonhole" bottoms. No materials which support combustion are used.

The "pigeonholes" are spaced a minimum of 12-inches edge to edge. The space between "pigeonholes" contains either expanded metal or metal rods to prevent the placing of unit containers between the "pigeonholes". The metal lumber is so arranged as to form a $1\frac{1}{2}$-inch high lip on the front of each pigeonhole. This lip prevents the unit container from sliding out of the "pigeonhole".

The racks on the ground floor are fastened to the walls so as to maintain vertical stability. The main support comes from the floor. The racks on the wooden cat walk use the walls for their main support. The cat walk is not necessary for support of these racks.

The size of the "pigeonhole" can be changed for different sized containers. The separation distance, which is 12-inches or more, can be increased when necessary to decrease the amount of neutron interaction.

Flooding is not considered credible except in the Depressed Area (waste disposal area). However, a minimum separation of 12-inches between "pigeonholes" is used to prevent any increased reactivity due to flooding.

5. Extruders (Exhibits III-10, III-11, Appendix B, Part III)

The extruders are used for semi-continuous forming of ceramic parts. The parts, as formed, are placed on saggers for processing in the furnaces described in 4 above. The bore dimensions for the different extruders are 4.8-inches I.D. by 22-inches long and 4.8-inches I.D. by 24-inches long.

Because of tooling design, a unit of special nuclear material cannot be completely extruded until a second unit has been added into the bore. Geometry control is effective for the material in the extruders. Administrative control along with procedural control is necessary for the placing of parts on the saggers.

Water flooding is not credible because of the location of the extruder. Flooding would not cause a criticality at the extruders because the extruders have a thick reflector already in the bore liner.

The extruders were operated for a six month period during the Tory II-C contract without any breakdown in the various types of control which were employed.

6. Dust Collectors (Exhibit III-12, Appendix B, Part III)

In glove boxes or velocity hoods where U-235 dust (wet or dry) is generated, the air is filtered by one of several filter systems employed. These are:

Mikro-Pulsaire Unit. The air and dust is pulled into a large chamber where the air passes through nine filter bags. Periodically a jet stream of air pulses in the opposite direction through the bags which knocks the dust off the bags down into a collection hopper. The hopper narrows down to a 4-3/4-inch I.D. by 16-inch cylinder.

The safety devices on this system include:

(a) four sight access ports for inspection,

(b) a vibrator on the hopper to insure that all dust moves into the 5-inch O.D. cylinder,

(c) two independently operating Bin-Dicators which would shut down the dust generating equipment if the dust stacked up in the geometrically unsafe portion of the hopper, and

(d) an air pressure differential sensing device across the filters which shuts down the dust generating equipment under high pressure differential conditions. This type condition would indicate material build-up on the filter bags.

Absolute Filters. These filters are used where minute quantities of dust are generated. As dust collects on the filters, the air flow rate decreases. The indicated (by differential pressure gage) air flow is below allowable operating conditions with only a few grams of material on the filter so that it is impossible to build up an unsafe quantity of material.

Mist Eliminators. The mist eliminators are located on the grinding equipment to keep grinding coolant and sludge out of the air ducts. The coolant drains back to the grinder or into a safe container (volume control). This system is also self regulating as the air flow is shut off as sludge or other material builds up on the mist eliminators.

7. Shipping Containers (Exhibits III-13, III-14, Appendix B, Part III)

The following types of packaging will be used for shipment:

Class II Package. This package does not need prior AEC approval for shipment. This container or group of containers for shipping not more than 160g U-235 per day will meet the requirements of 10 CFR 71.51.

4-inch Diameter by 40-inch Schedule 40 Steel Pipe. The pipe has welded end plates with a drain valve welded to the bottom end plate. The fill valve is bolted to the top end plate. The pipe is shipped in a 30-inch by 30-inch by 53-inch steel birdcage.

<u>5-inch Diameter by 30-inch Steel Pipe.</u> The pipe has screw-on end caps. The pipe is shipped or stored in the center of a 55-gallon drum (24-inch diameter by 34-inch high).

8. Shipments - Incoming and Outgoing

Incoming shipments will be received at the dock on the south side of the building. An empty trailer parked on the west end of the dock will be used to store the incoming material. The material from the carrier van will be unloaded one package at a time and moved to the empty trailer at the west end of the dock. The array shape will be the same in the receiving trailer as in the shipping van. Individual packages will then be moved to the sampling area and after analysis, into storage.

Outgoing shipments of completed product will be handled in the reverse manner. As individual packages are prepared for shipment, they will be moved to the same trailer at the west end of the dock provided that it is empty of other material. The material will then be loaded onto the shipping van when the whole shipment is ready.

APPENDIX B, PART I

Waste or scrap drums will be removed from the building by the west dock for storage inside the security fence. The drums will be stored in groups of fifty with 12-feet of separation between groups until the material is shipped to a recovery facility for reprocessing.

APPENDIX B, PART II

1. Oven (See Part I for description)

k Calculation

k was calculated for a bare slab 2.5-inches by 24-inches by 48-inches. The dimensions were arrived at in the following manner:

2.5-inches = 1.5-inch slab thickness plus 1-inch reflector savings.

24-inches = diameter of pan.

48-inches = 2 times the pan diameter because the two pans are effectively edge to edge (1/2-inch separation).

The two group method* was used for the k calculation. k is calculated to be 0.525. The criteria in 10 CFR 70.52 permits the solid angle of interaction to be:

9-10k, or

9-5.25 = 3.75 steradians

Solid Angle Determination

The two pans per shelf on each side of the center divider were converted into a 48-inch long slab with the same cross-sectional area as the two

*H. F. Henry, Studies in Nuclear Safety, K-1380, 1958

24-inch diameter pans.

The interaction for the two slabs directly above and below one of the center pans was calculated by the following equation:

$$\Omega = 4 \sin^{-1} \frac{(a/2) \cdot (b/2)}{\sqrt{(a/2)^2 + h^2} \cdot \sqrt{(b/2)^2 + h^2}}$$

where: a = 18.8-inches

b = 48-inches, and

h = 15.25-inches

Ω = 3.68 steradians

For the two slabs on the other side of the center divider, the interaction was calculated by the equation:

where a = 18.8-inches

b = 48-inches

$\cos\theta = h/q^2$

$q^2 = 4328\text{-inch}^2$

Ω = 0.0073 steradians

Ω (total) = .0073 + 3.68 = 3.69 steradians which meets the criteria of 10 CFR 70.52

2. Scrap Recovery System (See Part I for description)

a. k Determination

From 10 CFR 70.52 for a 5-inch or smaller cylinder, k is 0.58. The solid angle of interaction (Ω) then may be

$$\Omega = 9\text{-}10k, \text{ or}$$

$$9-5.8 = 3.2 \text{ steradians}$$

b. Interaction Calculation

$$\Omega = \frac{2d}{h} \sin \Theta$$

$d = 4$ inches

$h = 22$ inches

$$\sin \Theta = \frac{120}{(120^2 + 22^2)^{\frac{1}{2}}} = 0.983$$

$$\Omega = \frac{8}{22} \cdot 0.983 = 0.357 \text{ steradians}$$

For two cylinders interacting with the center cylinder, $2\Omega = 0.71$ steradians. This is well within the limits of 10 CFR 70.52.

APPENDIX B, PART II

3. Furnace - Hevi-Duty (See Part I for description)

This furnace is under geometry control. A buckling conversion from a 5-inch cylinder (reflected) to a 11-inch wide-infinite long slab is made. Although the opening of the furnace is 13-inches, only 11-inches is considered for U-235 materials. The other two inches are taken up as follows: one-half-inch on both sides of the sagger (1-inch total) as air space between the saggers and furnace walls, and one-half-inch for each of the walls on the saggers (1-inch total).

For the buckling calculation, the extrapolation length (δ) was obtained by equating the buckling for a 5-inch diameter infinite length cylinder to a 1.5-inch thick infinite length in two dimensions slab and solving for δ.

$$\left(\frac{Jo}{R+\delta}\right)^2 = B^2 = \left(\frac{\pi}{T+2\delta}\right)^2$$

$$\left(\frac{2.405}{6.35+\delta}\right)^2 = B^2 = \left(\frac{3.14}{3.81+2\delta}\right)^2$$

$$\delta = 6.45 \text{ cm}$$

Calculate the buckling for a 5-inch cylinder

0.0352

$$B^2 = \left(\frac{2.405}{6.35+6.45}\right)^2 = (.1878)^2 = 0.327 \text{ cm}^{-2}$$

6.35
6.45
12.80

APPENDIX B, PART II

Calculate the height of an 11-inch high slab using the above buckling

$$.0327 = \left(\frac{3.14}{27.94 + 12.9}\right)^2 + \left(\frac{3.14}{H + 12.9}\right)^2$$

$$.0327 = .0059 + \left(\frac{3.14}{H + 12.9}\right)^2$$

$$.0268 = \left(\frac{3.14}{H + 12.9}\right)^2$$

$$H + 12.9 = 19.19 \text{ cm}$$

$$H = 6.29 \text{ cm} = 2.48 \text{ inches}$$

The 2.48-inch by 11-inch slab was calculated from the minimum always safe cylinder diameter for compounds or solutions (See Figure 3 of 10 CFR 70) of any H/U-235 ratio. The slab is safe for any compound or solution of any H/U-235 ratio provided the criteria in Figure 13 of 10 CFR 70 is met.

For lower enrichments, the Allowance Factor of Figure 15 of 10 CFR 70 can be applied to this calculated slab thickness of 2.48-inches.

Furnace - Harper (See Part I for description)

The furnace is under geometry control and moderation control. All hydrogeneous moderator is removed in the Hevi-Duty furnace operation.

APPENDIX B, PART II

The chamber is 6-inches high by 7-1/4-inches wide which corresponds to a cross-sectional area of 42.5-in^2. This area is equivalent to a cylinder diameter of 7.44-inches which is well below the 10-inches for an H/U-235 ratio of 1 in 10 CFR 70, Figure 3. However, an H/U-235 ratio of 10 is assumed for safety. Figure 3 then limits the diameter of a cylinder to 6-inches. A buckling calculation from a 6-inch cylinder made to determine the height of a 7-1/4-inch by infinite length slab.

$$\left(\frac{2.405}{14.07}\right)^2 = B^2 = \left(\frac{3.14}{31.32}\right)^2 + \left(\frac{3.14}{H + 12.9}\right)^2$$

$$(.1709)^2 = B^2 = (.1002)^2 + \left(\frac{3.14}{H + 12.9}\right)^2$$

$$.0292 = .0100 + \left(\frac{3.14}{H + 12.9}\right)^2$$

$$.0192 = \left(\frac{3.14}{H + 12.9}\right)^2$$

$H + 12.9 \text{ cm} = 22.67 \text{ cm}$

$H = 9.77 \text{ cm} = 3.85$-inches

This 3.85-inch by 7.25-inch slab is safe for any U-235 compound with an H/U-235 ratio ≤ 10 and a U-235 density ≤ 2 g/cm^3. (Density reference - TID-7016 Rev I, Figure 3)

For lower enrichments, the Allowance Factor of Figure 15 of 10 CFR 70 can be applied to the calculated slab height of 3.85-inches.

4. Storage Racks

10# Fruit Can Storage (See Part I for description)

a. k Determination

For a mass limited container, $k \doteq 0.65$, which permits a solid angle of interaction (Ω) of

$9-10k$, or

$\Omega = 9-6.5 \doteq 2.5$ steradians

(Ref. 10 CFR 70.52)

b. Interaction Calculation

This calculation is for "in process" storage racks where there is no interaction between racks except in corners of an area where two racks can be located at right angles to each other. It should be noted units in two storage racks at right angles to each other are spaced a minimum of 30-inches edge to edge. Material stored in this container in the vault will be handled separately.

The 7-inch diameter by 9-inch high cans are stored in a horizontal position (axis perpendicular to the vertical plane of the rack). The solid angle of interaction was calculated by the equation

$$\Omega = \frac{2d}{h} \sin \Theta$$

c. This calculation is for vault storage (cubic array)

From Figure 1, TID-7016, Rev 1, the minimum critical mass for a minimally reflected unit is 600g. The 350g unit which is to be stored in a can with less than 1/8-inch steel reflector so that it can be considered to have minimal reflection. The array is considered fully reflected.

The 350g unit is stored in a 20-inch cubic array. This is equivalent to storing a 600g unit in a $\left(\sqrt[3]{20 \times 20 \times 20 \times \frac{600}{350}}\right)$ 24-inch cubic array. From Figure 22, TID-7016, Rev 1, 75 units of 600g each are allowed per array. The 75 units of 600g each are equivalent to ($75 \text{ units} \times \frac{600}{350} =$) 128 units.

128 units of U-235 with 350g per unit can be stored in each bay of the vault. Additional safety is realized because the units are

actually stored in two (1 x 8 x 8) planes separated by 30-inches.

The calculation for all interaction in a 5 high by n long unit array yields $\Omega = 1.75$ steradians which meets the criteria in 10 CFR 70.52.

30# Fruit Can Storage (See Part I for description)

a. k determination

As for the 10# Fruit Can (above), $k = 0.65$ and $\Omega = 2.5$

b. Interaction calculation

This calculation is for "in process" storage. Plans do not call for using this container for vault storage.

The 10-inch diameter $12\frac{1}{2}$-inch can is stored with its axis perpendicular to vertical plane of the rack. The solid angle of interaction was calculated by the equation

$$\Omega = \frac{2d}{h} \sin \Theta$$

The results of calculations for a (5 high by n long) unit array yield a value of 2.5 steradians which meets the criteria of 10 CFR 70.52.

5. Extruders (See Part I for description)

The extruders with 4.8-inch I.D. by 22 or 24-inch long bores are under a combination of geometry and unit control. The bores are safe for any U-235 compound which meets the maximum density limits as a function of degree of moderation in 10 CFR 70, Figure 13.

If the density exceeds the safe value in Figure 13, the safe cylinder diameter will be adjusted in accordance with 10 CFR 70.43. If the adjusted cylinder diameter is less than 4.8-inches, mass limits will be employed as the primary means of control.

No calculations are presented at this time because there is no process requirement at the present time for U-235 compounds not in accord with Figure 13 of 10 CFR 70.

6. Dust Collectors, Mikro-Pulsaire (See Part I for description)

The collection hopper is a 4-3/4-inch I.D. by 16-inch cylinder. It has a capacity of 4.65 l which is safe for any U-235 compound. 4.8 l is the maximum always safe volume. (Ref. TID-7016, Rev 1)

7. Shipping Containers (See Part I for description)

4-inch Diameter by 40-inch Pipe

a. k Determination

This pipe is limited to less than 2000g of U-235 in solution ($H/U\text{-}235 > 20$)

From 10 CFR 70.52, for a 5-inch or smaller cylinder, k is 0.58. The solid angle of interaction is calculated to be:

$$\Omega = 9-10k = 9-5.8 = 3.2 \text{ steradian}$$

b. Interaction Calculation

For 4-inch I.D. by 40-inch cylinders in a square array on 30-inch centers, the solid angle of interaction is the sum of the solid angles of interaction between the center cylinder and all unshielded cylinders in the array. The solid angle of interaction for two cylinders is calculated by the formula:

$$\Omega = \frac{2d}{h} \sin \Theta$$

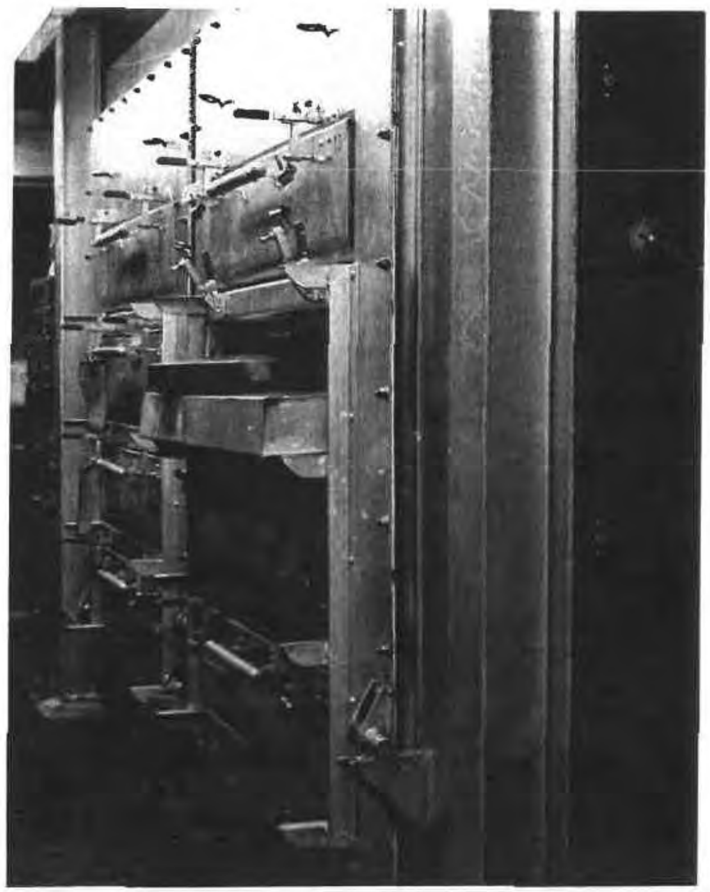
which is taken from 10 CFR 70.52. A central unit in an infinite planar array has a solid angle of about 2.2 steradians which is below the limit as defined in the k determination.

5-inch Diameter by 30-inch Pipe

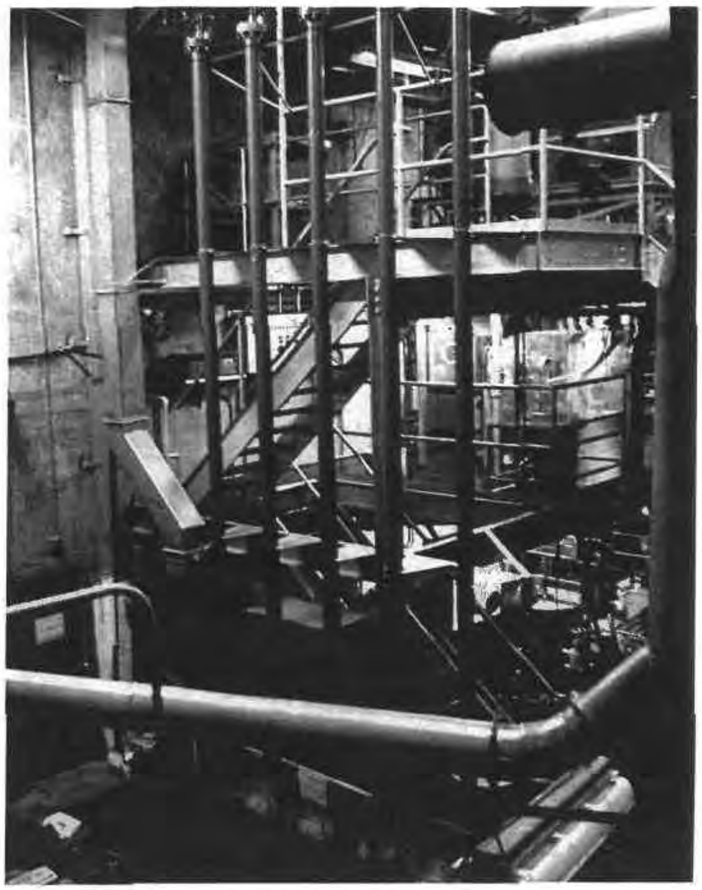
This pipe is limited to less than 1000g of U-235 as waste compounds held for recovery. The drums will be stored in groups of 50 with twelve feet of separation between groups per the criteria as developed by Ketzlack in NAA-SR-MEMO-6415, or in a plane array with 1.5 feet edge to edge separation per criteria of Proposed 10 CFR 70.

8. Shipments - Incoming and Outgoing

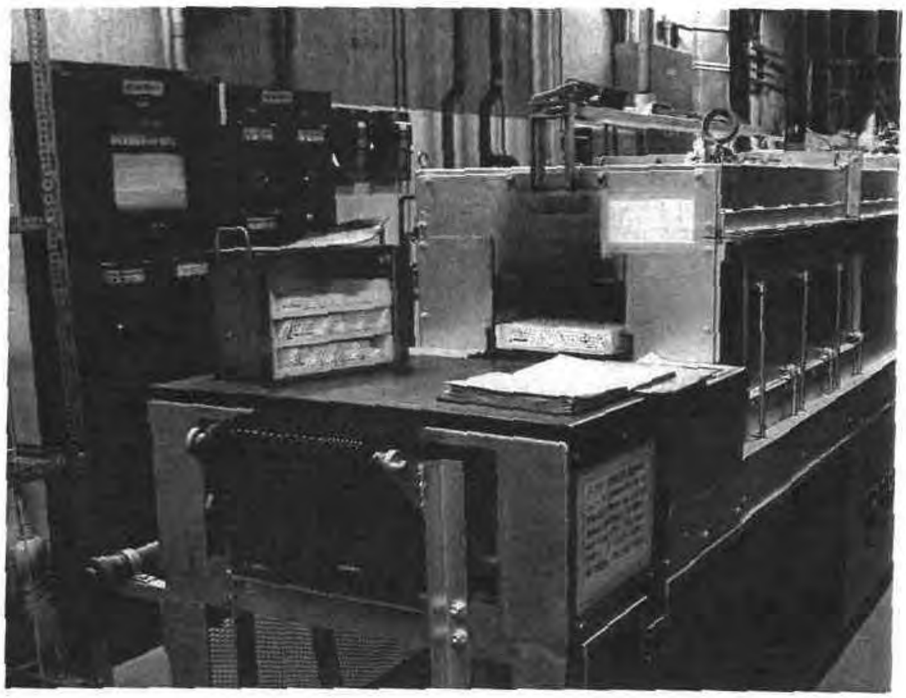
The nuclear safety evaluation of shipping containers in arrays is given in Section 7, the preceding section.



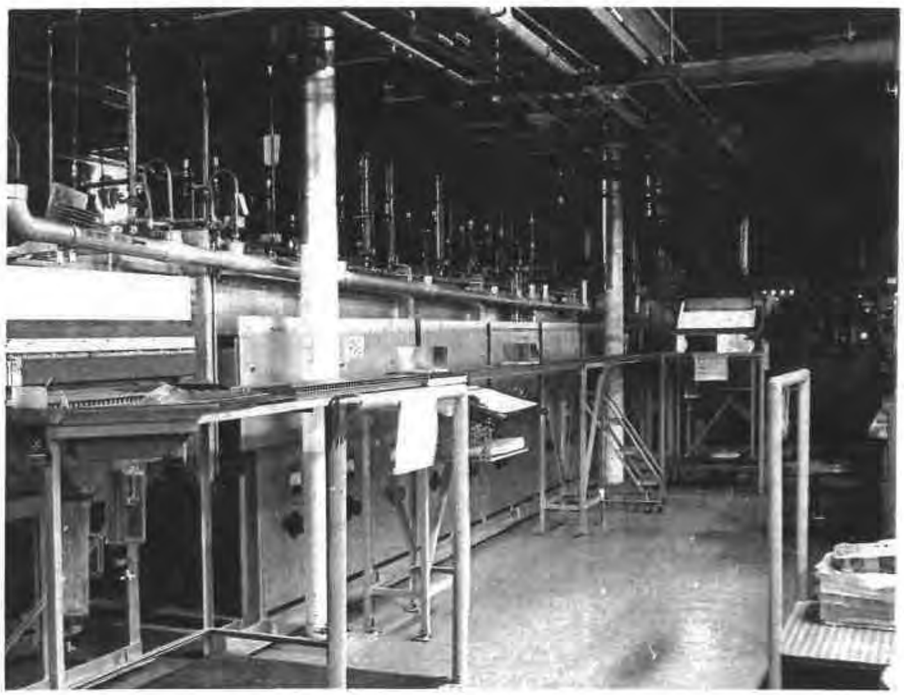
One of the eight doors of the oven is open to show one of the pans which has been partially withdrawn.



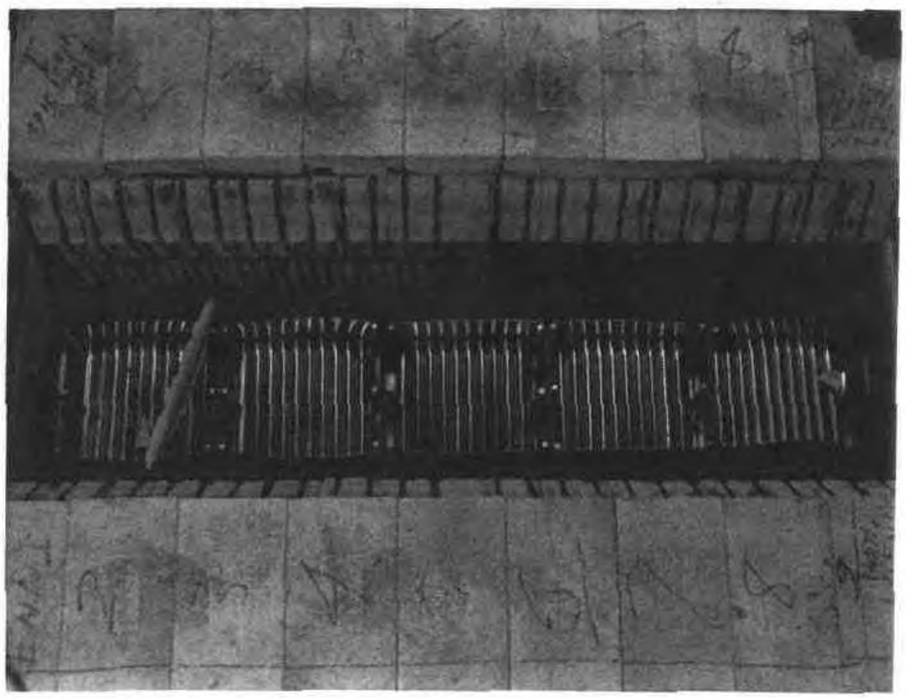
Note the expanded metal guard one
foot in front of the tanks on the
ground level.



View shows a tote box for handling one unit of material on up to four saggers at the loading end of the Hevi-Duty furnace.



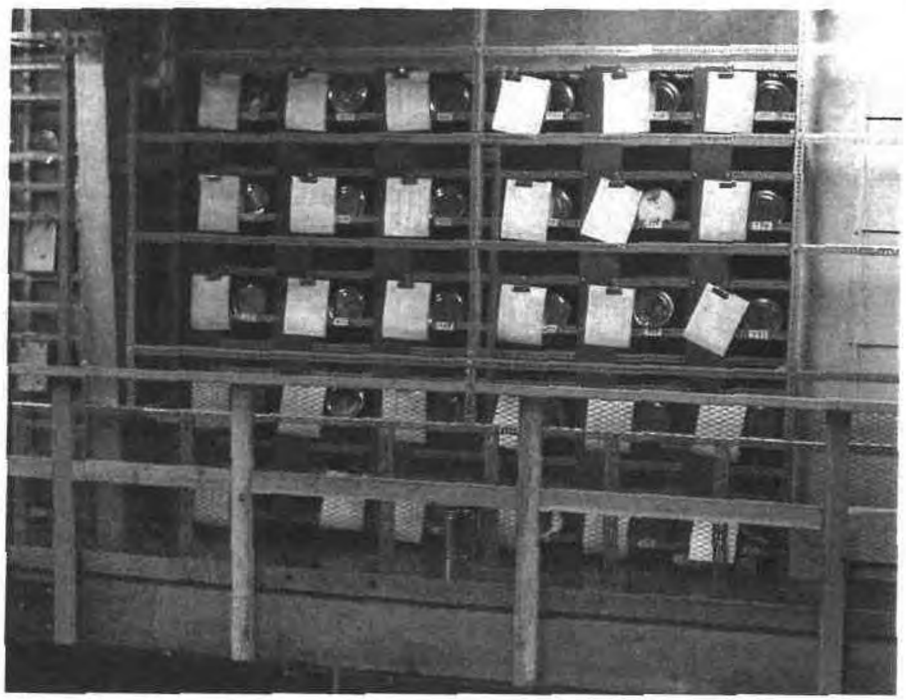
The sagger unloading station (in left foreground) and conveyor are shown in front of the furnace. The saggers are kept on the conveyor until they are unloaded. Only two loaded saggers are out of the furnace at any given time.



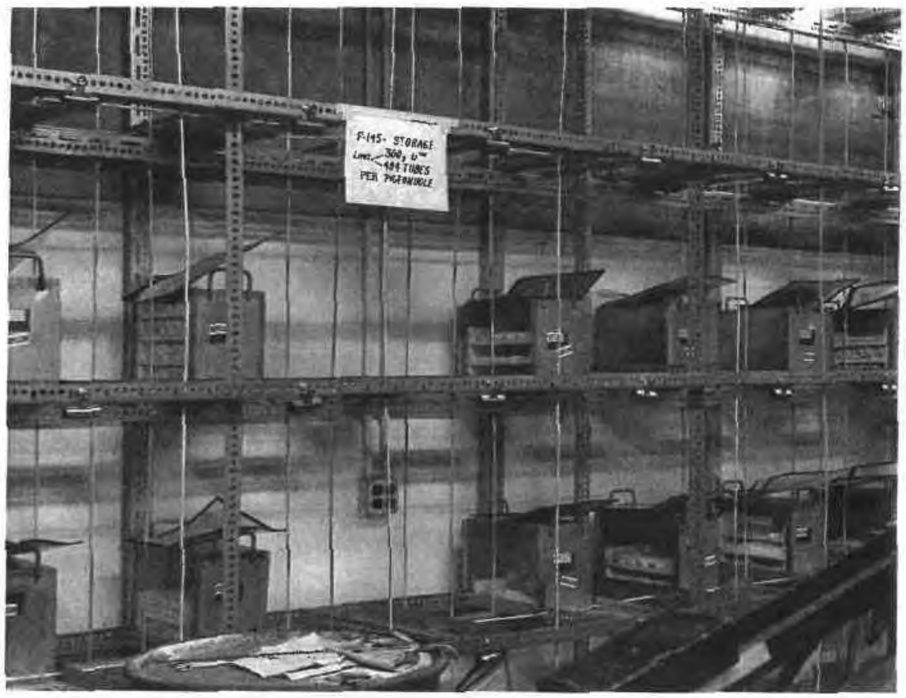
The top of the furnace has been
removed exposing the parallapiped
arrangement of saggers in the
furnace.

This is a view of the main in-process storage area. Other racks are located on the production floor with the production equipment.

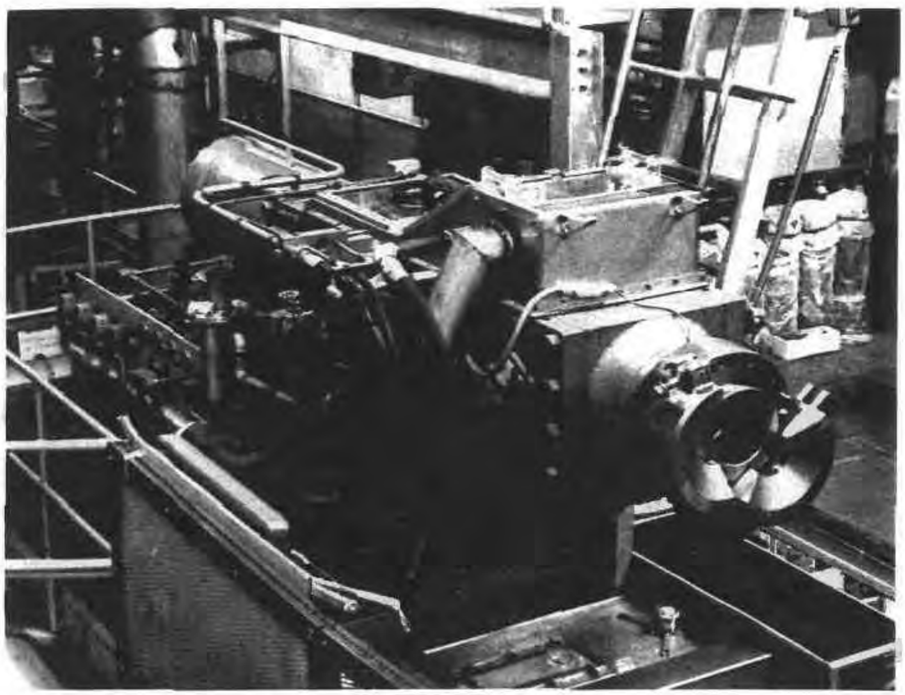
Close-up view of 30# Fruit Can Storage
Rack for inprocess storage on the
catwalk.



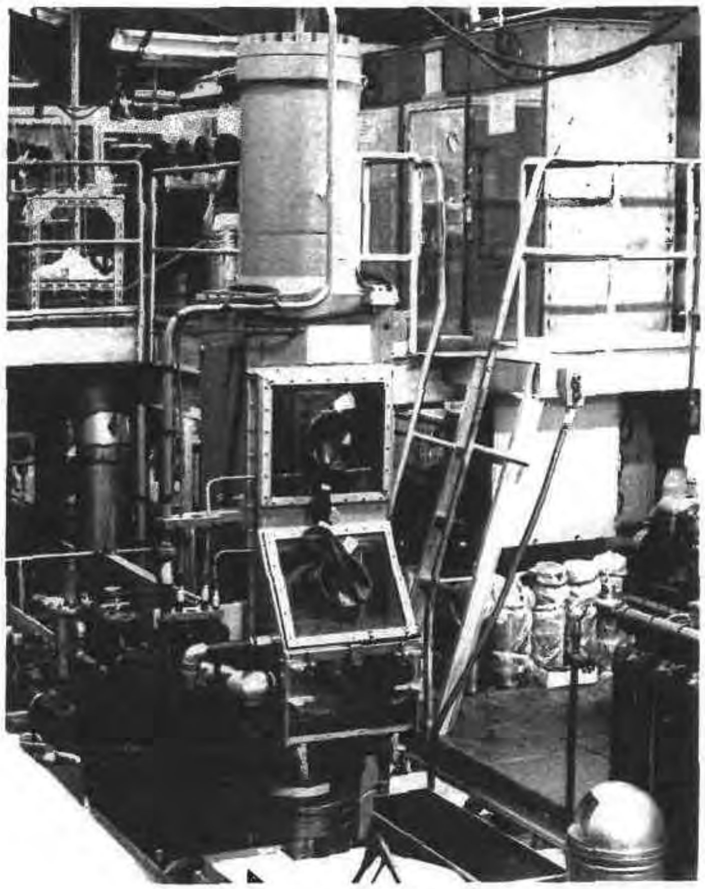
Close-up view of 10# Fruit Can Storage
Rack for inprocess storage on the
catwalk.



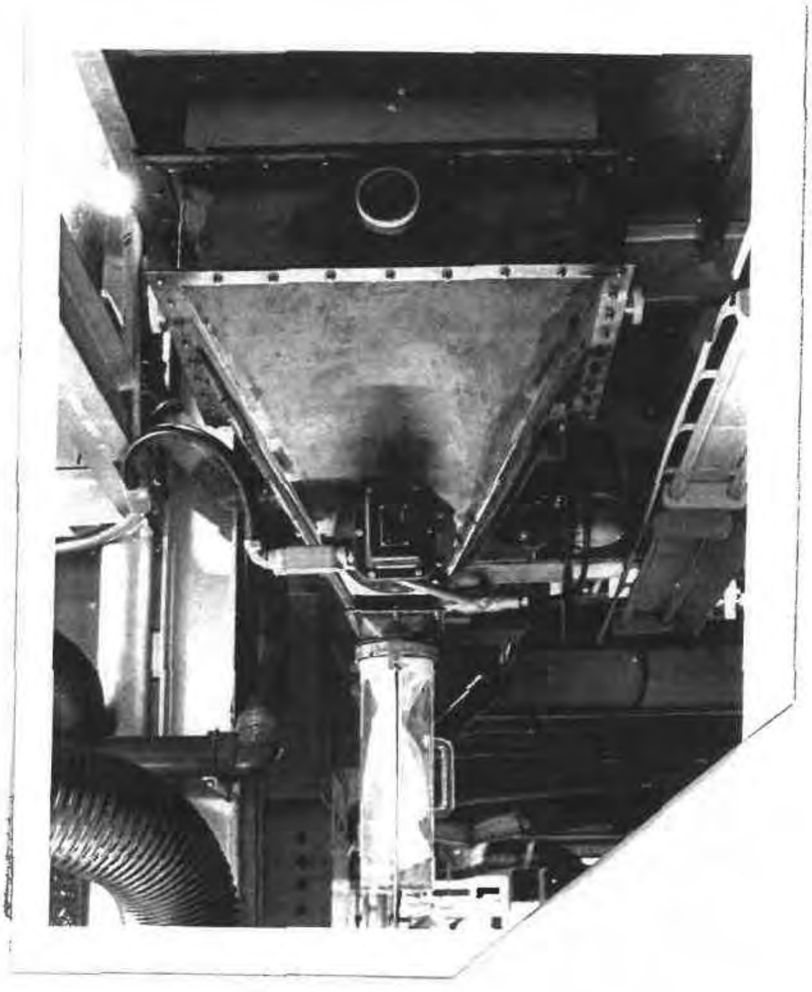
Close-up view of tote box storage
rack in the furnace room.



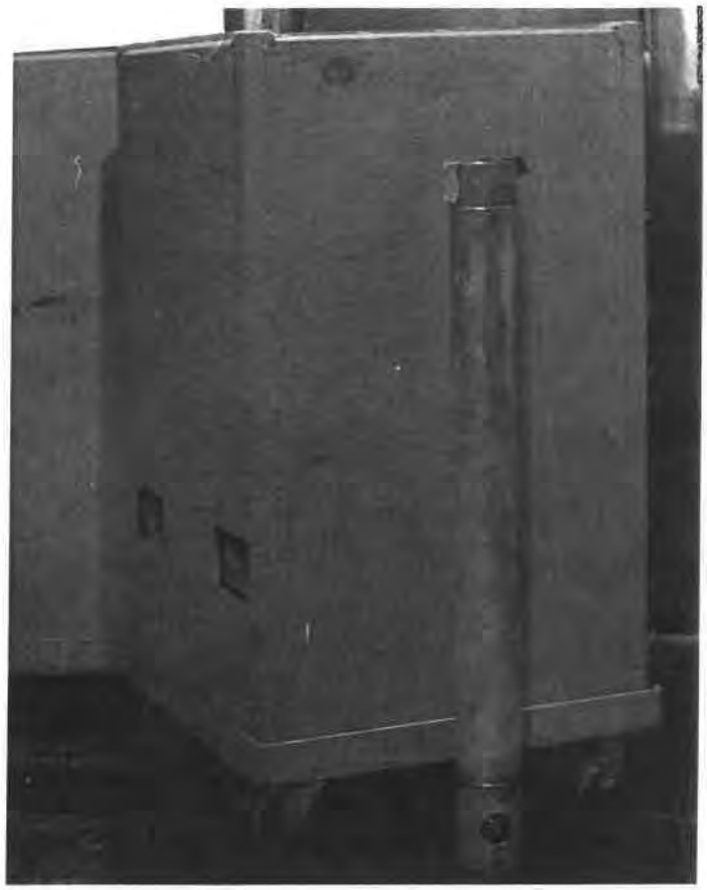
The extruder has been moved to the
horizontal position. The tooling
has been removed to expose the
4.8-inch diameter bore.



The extruder, in the vertical position,
shows details of the loading glove box.



The view shows the hopper as it narrows down to the 4-3/4-inch diameter cylinder, the bin-dicator (level indicator) on the front of the hopper, and the vibrator mounted on the side of the hopper.



The sections welded on the top and bottom of the pipe are to protect the valves for filling and emptying the pipe. The birdcage can be transported by means of a dolly or by a forklift. The thin steel sheet on the sides and top of the birdcage are not structural members but are intended to keep unwanted materials from entering the birdcage framework during shipment.

View shows 5-inch pipe and method of locating the pipe in the center of the drum.

Coors Porcelain Company, Source Material License SMB-746

License Active: March 16, 1964 - March 31, 1967

Location: Energy Products Division
Coors Porcelain Company
Fuel Element/Coors Research Building
600 Ninth Street
Golden, Colorado 80401

Activity: A maximum quantity of 30,000 pounds of uranium and thorium was authorized for fuel element fabrication as well as for research, development, manufacturing, and testing activities. The licensee apparently fabricated ceramic fuel elements for an experimental nuclear powered jet engine.

ORNL Concerns:

1. There was no verifiable decontamination of the site at closeout.

2. The disposition information in the docket files only accounts for less than 200 pounds of source material. If the licensee possessed any amount near the authorized limit, then a large amount of source material cannot be accounted for.

3. There was no closeout survey or final AEC/NRC inspection of the facility.

Region IV Remarks:

1. On December 19, 1963, Coors Porcelain Company submitted an application to the AEC for a source material license and a special nuclear material license. Source Material license SMB-746, Docket 40-7096, was issued on March 16, 1964.

2. A separate classified file was developed for the special nuclear materials license (Docket 70-814). Since information in the SMB-746 file was limited in scope on the details of the facility, Docket File 70-814 was located in the NRC archives and was reviewed. According to information provided in the 70-814 docket file, the licensee was not issued a special nuclear materials license. In August 1964, the licensee requested a withdrawal of the special nuclear material license application. In conclusion, this docket file provided little information about the final disposition of the source material.

3. License SMB-746 expired on March 31, 1967. Just prior to the expiration, Coors sent a letter to the AEC stating that all source material previously authorized

under the license would be possessed in accordance with the general license requirements of 10 CFR 40.22. This regulation authorized the possession and use of source material not to exceed a 15 pound limit.

4. In a February 16, 1968, letter to the AEC, the licensee informed the Commission that roughly 100 pounds of source material was "found" at the facility. This material was apparently disposed of at the Dow plant at Rocky Flats around March 1, 1968.

5. A review of the Coors file at the State of Colorado offices in February 1994 identified transcripts of an October 1992 television station investigation series concerning allegations that: (1) hundreds of pounds of weapons grade uranium from Coors Porcelain apparently had been smuggled out of the country, (2) some of the radioactive material previously transferred to Rocky Flats was leaching from the storage containers, and (3) some contaminated ventilation duct work had been buried in a mine shaft. The State was aware of the allegations but had decided not to follow up on the investigative series findings. A copy of the transcripts was provided to the Region IV Allegations Coordinator.

6. During April 1994, the Department of Energy was contacted for any information that they may have about Coors Porcelain. The DOE responded with information that they had concluded in 1987 that this facility had been adequately decontaminated and that no Formerly Utilized Sites Remedial Action Program (FUSRAP) followup action was required.

7. On May 20, 1994, an NRC inspector visited the Coors Ceramic Company (name had changed). The inspector was accompanied by a State of Colorado inspector. According to the Coors Ceramic representative, the company stopped fabricating fuel elements during the 1960's. The building was then used for manufacturing products made with beryllium for a period of time. Morrison-Knudsen was contracted to decontaminate the building of all hazardous materials, primarily beryllium. Following decontamination of the facility, the building was demolished around 1986.

8. During the May 20, 1994, visit to Coors Ceramic, the Environmental Health and Safety Administrator provided the NRC inspector with a number of documents, including an August 19, 1985, letter from Morrison-

Knudsen Engineers, Inc., stating that the building had been sampled for radioactive materials. The letter stated that the sample results were below federal and state guidelines for radioactive materials in non-restricted areas. Also, a contractor surveyed the former building area two days before the inspector arrived at the facility and no radioactive readings above background were observed by the contractor.

9. During the May 1994 visit, the Environmental Health and Safety Administrator for Coors Ceramic took the NRC and State inspectors on a tour of the facility. The Fuel Element Building had been demolished. The area where the building was previously located was now a parking lot. No radioactive exposure rates above background levels were observed during the tour of the area where the building had been situated.

10. During the 1970's, Coors Porcelain disposed of selected materials in the Glencoe Mine in Jefferson County, Colorado. This mine was sealed in 1979. In part to public pressure, Coors Ceramic opened the mine in July 1994 and began removing the hazardous materials from the mine. One ventilation filter was identified with radioactive contamination. The filter measured 152 microRoentgen/hour on contact. This filter may have came from the Fuel Element Building prior to the mine being sealed. No other radioactive material was found in the mine. To compound problems, the filter was sampled and was found to contain PCB's. At the time of this memorandum, the former licensee planned to dispose of the material at an authorized DOE facility in the near future.

Regional Recommendation: Region IV recommends removal of this site from the Terminated Site List. The Fuel Element building was demolished and no residual radioactive materials were identified during a brief visit to the facility. The former licensee committed in writing to inform the NRC when the contaminated filter has been properly disposed of. The investigative series allegations that material had been shipped overseas or that material at Rocky Flats was leaching into the environment were not investigated; however, these problems are not the responsibility of Coors Ceramics.