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KERR-McGEE CORPORATION

KERR-McGEE BUILDING • OKLAHOMA CITY, OKLAHOMA 73102

Regulatory Suppl File Cy.

February 12, 1969

RECEIVED FEB 13 1969 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

Dr. John A. McBride, Director
Division of Materials Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. McBride:

Subject: Renewal of License SNM-928, Docket 70-925

This will transmit revised pages for incorporation in the pending application for renewal of License SNM-928 issued to Kerr-McGee Corporation. The original renewal application was dated August 30, 1968. An additional copy of the complete application is also furnished as requested during my visit on December 12.

We have endeavored to incorporate the changes and additions which have been suggested during discussions with members of your staff. A request for approval of a new transport package (Model KM-1) has been added. The attachment to this letter lists the revised or new pages with a description of the changes involved in this Amendment No. 1 to the consolidated renewal application.

It is requested that Drawing NPD-107-201 (Rev. 1) be withheld from public disclosure, pursuant to Section 2.790(b) of the Commission's Rules of Practice, 10CFR Part 2, for the same reasons given in our request of August 30, 1968.

We anticipate that we have been responsive to the needed changes for license renewal, and appreciate your consideration of the amended application. Please advise if you have any questions regarding this submission.

DOCKETED USAEC FEB 13 1969 REGULATORY MAIL SECTION DOCKET CLERK

Very truly yours,

G. E. Wuller

G. E. Wuller
Nuclear Division Staff Engineer
Licensing and Regulation

GEW:cjp
Enclosures:
Amendment #1, SNM-928 Renewal Application (7 copies)
One complete renewal application SNM-928

ACKNOWLEDGED

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RENEWAL APPLICATION - LICENSE SNM-928, DOCKET 70-925

Subject: Summary of Changes - Amendment No. 1 February 12, 1969

New Pages (2/12/69)	Section Amended and Reason	Replaces Pages (8/30/68)
Drawing NPD-107-201	Rev. 1, 12/20/68 - Added Fuel Rod Fabrication Layout (Process Area F)	8/9/68 issue
i,ii,iv	Table of Contents Revised	i,ii,iv
1-3	1.6f - New Proposed Activity - Manufacture of Fuel Rods	1-3
2-3	2.3.3 Personnel Title Change	2-3
2-8	2.3.13 Personnel Termination (Information Deleted)	2-8
4-1 & 4-1.1	4.1 Revised List of Monitoring Equipment	4-1
4-6 to 4-7.1	4.5.1 Revised Criticality Alarm System	4-6 & 4-7
5-19	5.8.3b - Defined Units for k_{eff}	5-19
5-21 & 5-22	5.9 New Section - Laboratory Criticality Control	New Pages
5-23	5.10 New Section - Calculation for Nominal Reflection	New Page
11-1 to 11-4	11.0 New Section - Added Fuel Rod Manufacture	New Pages
12-1	12.1-6 Added Package Model KM-1	12-1
12-25	12-3.2 Revised Class II Transport Index	12-25
12-33	12-4.2a Revised Class II Transport Index	12-33
12-34	12-4.2 Revised Class II Transport Index	12-34

Summary of Changes

APPENDIX A

New Pages (2/12/69)	Section Amended and Reason	Replaces Pages (8/30/68)
i,ii	Table of Contents Revised	i,ii
1-1	1.2.1-1b Added Fuel Rod Manufacture	1-1
2-1 to 2-5	2.0 Organization, etc. - Added Personnel Qualifications	2-1 to 2-3
3-1.0 to 3-1.2	3.0 Radiological Protection Program Revised for Definitive Requirements	3-1
3-3 & 3-4	" " "	3-3 & 3-4
3-5.0 to 3-5.2	" " "	3-5
4-5	4.7-2 Specified Reactivity Limit for 8" Concrete Isolation	4-5
4-9 & 4-10	4.10 New Section - Administrative Requirements	4-9
5-1	5.0 Added Package Model No. KM-1	5-1
5-5	3.3.1 Revised Class II Transport Index	5-5
5-6	4.2.2b Corrected Table of Enrichments Versus Masses, and 4.3.1 Revised Class II Transport Index	5-6
5-8	5.0.6 New Section - Package Model KM-1	New Page

APPENDIX D

1 to 12	Appendix D - Bases for License Conditions	New

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Attachments:

1.5 Material Description

The renewal of License SNM-928 is requested to cover the receipt, possession and processing of enriched uranium at any assay of the isotope U-235. A maximum possession quantity of 3,000 kilograms of contained U-235 is requested.

The special nuclear materials will be received as enriched UF_6 or other uranium compounds or mixtures including uranium scrap materials and will be processed and converted to compounds of uranium suitable for nuclear fuels. The enriched uranium material may be in any of a number of physical forms. The more commonly handled forms will be ceramic powders, fused crystals, pelletized powders, solutions, intermediate salts, and metal. It is anticipated that the type, quantity and assay of special nuclear material will vary considerably from time to time. It is estimated that an average of 15,000 kilograms of contained U-235 will be processed annually.

No request for possession of plutonium or uranium-233 is included in this application.

1.6 Summary Description of Proposed Activities

The general plan for the licensed activity principally consists of six major process areas, namely:

a. Manufacture of Low Enriched UO_2 Powder and Pellets (Process Area A)

b. Pelletizing of Enriched Uranium Compounds (Process Area B)

c. Production of Low Enriched Uranium Compounds (Process Area C)

d. Recovery of Scrap (Process Area D)

e. Processing of High Enriched Uranium Compounds (Process Area E)

f. Manufacture of Nuclear Reactor Fuel Rods containing Enriched UO_2 (Process Area F)

Each of the above process areas has problems unique to their operation. Certain criteria regarding health and nuclear safety in addition to the generally applicable procedures, are separately presented for each process area for clarity and ease of evaluation. Additionally, the Cimarron Facility Laboratory provides analytical and quality control services in the areas of chemical and physical testing for the entire facility.

Manager, Engineering Development, Hanford Laboratory, General Electric Company, for eleven years. Fuel production, reactor operation and chemical separations, including direction of plutonium recycle test reactors project concept, design, construction and operation and direction of plutonium fabrication pilot plant project.

Associate Manager, HTGR Division, General Atomic/General Dynamics, for four years. HTGR fuels, development and program management, including research and development, design, reactor construction and fuel fabrication for Peach Bottom HTGR, Experimental Beryllium Oxide Reactor and Colorado Public Service HTGR.

Dr. Fryar reports to the President of Kerr-McGee Corporation and is responsible for all Nuclear Division activities, including development, marketing, engineering and manufacturing.

2.3.2 William J. Shelley - Manager, Manufacturing

Education - Master of Science

Experience - Employed by Mallinkrodt Chemical Works from 1949 to 1967. Served as Project Engineer with assignments in construction, research and development, and plant startup; Administrative Assistant responsible for the operation of division purchasing, office services, material accountability and classification sections; Production Control Manager responsible for production scheduling, production reporting and material accountability functions; Director of Administration; and Assistant Division Manager.

Mr. Shelley was General Manager of the Mallinckrodt Uranium Division for six years, in which capacity he was responsible to the USAEC for the operation of its $50,000,000 chemical refining and metal fabrication facilities located at Weldon Spring, Missouri. Mr. Shelley served for three years as Vice President of Mallinckrodt Chemical Works and as a member of the Corporate Operating Committee. Mr. Shelley joined Kerr-McGee in July, 1967. He reports to the Vice President, Nuclear Division, and is responsible for all manufacturing operations of the Nuclear Division.

2.3.3 Dr. F. K. Pittman - *Manager, Technical Services*

Education - Ph.D.

Experience - Instructor of Inorganic and Analytical Chemistry for Massachusetts Institute of Technology for two years.

Research Chemist for one year; followed by four years employment by the Los Alamos Scientific Laboratories.

2.3.12 J.R. Davenport - Supervisor, Production

Education - Bachelor of Science, Chemistry and Biology

Experience - Employed eight years by Allied Chemical Corporation. Mr. Davenport joined Allied as an Analytical Chemist, and after six months was promoted to Production Chemist. In this capacity, he was involved in the manufacture of fluorine and sulfur hexafluoride. He held this position for two and one-half years and was promoted to Production Foreman in the manufacture of uranium hexafluoride.

Joined Kerr-McGee in 1967 as Supervisor, Production, Cimarron Facility, and reports to the Facility Manager.

2.3.13 *Information deleted*

2.3.14 A.L. Dungey, Engineer

Education - Bachelor of Science, Mechanical Engineering
Bachelor of Science, Business Administration

Experience - Employed as Plant Engineer for American Salt Corporation for two years. Mr. Dungey was responsible for total maintenance and engineering of the American Salt plant in Lyons, Kansas.

Employed by R.W. Booker & Associates, consulting engineering firm in St. Louis, as Mechanical Engineer for three years. Worked on projects for Monsanto Chemical Company, revising existing facilities and designing new equipment.

Worked for Union Carbide Corporation at the Atomic Energy Commission Paducah Gaseous Diffusion Plant as Maintenance Engineer for one and one-half years. He was responsible

4.0 RADIOLOGICAL PROTECTION

This section describes the programs, equipment and techniques employed at the Cimarron Facility for protection against radiation while conducting the authorized activities involving special nuclear material.

4.1 Detection and Monitoring Equipment - The following equipment is available and used at the Cimarron Facility to assure that the established radiation limits are not exceeded:

*Type-Use	Model	Radiation Detected	Range
Ionization Chamber-Radiation Detection			
Dose Rate	Eberline GADORA-1	γ	1-1000 R/hr
Dose Rate (2)	Eberline Atometer 6112	β-γ	0-15 R/hr
Dose Rate	NUCOR CS-40A	β-γ	0-50 R/hr
Scintillation-Radiation Detection			
Contamination	Ludlum - 12	β-γ	0-500,000 cpm
Control (2)	Eberline PAC-1SA	α	0-2 million cpm
Control	Eberline RM-3C	α	0-50,000 cpm
Criticality monitors (4)	NMC GA-2	α	0-50 mr/hr
Geiger-Mueller Tube Radiation Detection			
Dose Rate	Eberline 500B	β-γ	0-2 R/hr
Dose Rate	Electro Neutronics CDV-70	γ	0-5 R/hr
Dose Rate	Electro Neutronics CDV-700	β-γ	0-50 mr/hr
Proportional Counters			
Air Sample Counter	NMC PC-3T	β-γ	-
Utility Counter	NMC PC-3A	α-β-γ	-
Pulse Height Analyser	NMC PHA-1CA	γ	-
Air Velocity Meter			
Airflow	Hastings Velometer B-16A	-	0-20,000 LFM
Portable Battery Air Samplers			
Air Sampling (2)	Shop made	-	0-15 LPM
Air Sampling	Gelman 15003	-	0-18 LPM
Portable Line Air Samplers			
Air Sampling (5)	Gelman 13400	-	0-2.9 CFM
Air Sampling (4)	Staplex TFI-A	-	0-20 CFM
Leak Detector - for hydrogen and natural gas		*	
Davis-Vapo Tester			

All survey and sampling equipment is inspected and calibrated at intervals sufficient to assure reliable operation.

4.2 Contamination Control

To guard against contamination, protective clothing is worn by all personnel throughout the plant area and laboratories. Use of the company-provided clothing protects the wearer from contaminating himself and his own clothing and helps confine radioactive contaminants to the work areas.

The use of the regulated change rooms with locker-shower facilities makes it possible to maintain clean areas and have an effective control of radioactive contamination. Signs are posted at the entrance to the process area specifying the safety clothing and personnel monitoring device requirements for entry. Proper radiation caution signs are also conspicuously posted at the appropriate entrances.

to the Cimarron River. Analyses of the samples determines the concentrations of radioactivity in the effluent and are used to demonstrate compliance with 10CFR20.106(a). Calculations and experience have shown that the concentration levels of radioactivity in the total plant effluent are several orders of magnitude below the allowable limits of 10CFR20, Appendix B, Table II without considerations of river dilution.

The generation of any sizeable quantities of uranium being solid wastes is not expected for the Cimarron Facility. In the event that solid wastes are to be discarded rather than held in storage, any disposal will be in accordance with the provisions of 10CFR20.

4.5 Emergency Planning and Control

*4.5.1 Monitor Alarm System

As required by 10CFR70.24, a monitoring system with an audible alarm is maintained at the Cimarron Facility to warn personnel in the event a condition of accidental criticality occurs. The criticality alarm system consists of five gamma radiation sensitive stations which will energize an air horn alarm system. The alarm is clearly audible throughout the building and surrounding area. Four locations of the monitoring and alarm devices are shown on Drawing NPD-107-201(Rev.1). Another station not shown is installed on the inside south wall of Building 2. A gamma sensitive station is within 120 feet of every location where special nuclear material is handled, used or stored or at such lesser distances to adequately compensate for intervening shielding.

Each of the five gamma radiation sensitive stations consists of two instruments which are connected in coincidence into the alarm circuit. That is, a high level signal is required from both instruments of any station to sound an alarm. However, if one of the two instruments has failed (which produces a low level signal) the other instrument will cause an alarm to sound on a high radiation level.

The output of each of the ten instruments is monitored and recorded by a multipoint recorder located in the office area. Alarm lights are attached to this recorder which provide a visual signal for low alarm (yellow) and high alarm (red) from each instrument. This recorder also provides a permanent record of the signal from each instrument. This recorder is visable from outside the building through the glass front door. It can be read before re-entering the building after an evacuation.

Each instrument contains a detector, gamma source, meter high and low alarm lights, a memory circuit with a manual reset high alarm light, amplifier and relays for connecting external alarm devices. The radiation detectors are gamma sensors manufactured by Nuclear Measurements Corporation, Model GA-2A, which have a range of 0 to 50 mr/hr. These instruments have two adjustable pointers by which high and low level set points may be established. A needle continuously indicates the radiation level. A built-in radiation source insures that the needle indicates a radiation level of at least 0.2 mr/hr. at all times while the instrument is functioning properly.

A low level set point of 0.1 mr/hr. is used. If the needle drops to the low set point, a large yellow light in the upper left corner of the instrument case will light indicating that the instrument is out of order. The high level set point is maintained at 6 mr/hr. If the radiation level should exceed 6 mr/hr., the alarm relay will operate producing an alarm signal to the coincidence circuit and the large red light in the upper right corner of the instrument case will light up, a small red light will also light up and stay lighted until manually reset at the instrument. A fourth light on the instrument case, a small white pilot light, will remain on as long as power is supplied to the gamma detector. The meter will reach the 6 mr/hr. level within 0.2 of a second after the detector is exposed to a 20 mr/hr. source and will activate the alarm relay. The alarms will automatically sound within 0.1 of a second after the 6 mr/hr. set point is reached. Therefore, the alarms will sound within 0.3 of a second after any one of the stations is exposed to a 20 mr/hr. source.

The alarm system is powered by a 1.0 KVA, 115V, 60Hz solid state reverse transfer static inverter power system. It is normally connected to commercial power. On a power failure, the load is carried by nickel-cadium batteries until the emergency power engine comes to full speed or for a maximum of at least 1/2 hour.

The alarm air horn is connected through a "fail open" solenoid valve to an air tank which is pressurized through a check valve by dry instrument air and by nitrogen. Should the power to the solenoid valve be accidentally interrupted the alarm horn will sound.*

4.5.2 Testing and Training

Once during each shift the gamma alarm system recorder is inspected by the supervisor in charge. He dates and initials the chart at that time. This inspection is to assure that the system is operating properly and that the measured radiation does not reflect a significant increase over the normal background level.

The built-in low level source on each detector provides a calibration check at one point. A portable source is used to check the instruments at higher levels. The detectors are calibrated at three-month intervals. The alarm system is completely tested at least quarterly. Instructional meetings are held for all personnel to familiarize them with the criticality alarm system and the procedure for evacuating the plant in the event of a radiation incident. Practice evacuation drills are held at three-month intervals.

4.5.3 Evacuation and Emergency Procedure

Prompt evacuation of the plant is made if the alarms sound for any reason other than planned testing. Criticality evaluation and emergency procedures for supervisory and operating personnel are established and appropriate personal copies distributed to all individuals.

On a day shift, the Cimarron Facility Manager is responsible for coordinating the emergency actions. The Production Superintendent will assume the responsibility in the absence of the Facility Manager. If both the Facility Manager and Production Superintendent are absent at the time of a criticality incident, the Shift Supervisor has the responsibility for conducting the preplanned emergency actions and taking all necessary precautions to protect health and minimize danger to life and property.

(c) For units of safe masses based on moderation control, analytical verification of the hydrogen content will be obtained and the H/U-235 calculated prior to placing the unit in the storage array.

(d) Units stored using conditions of a specified H/U-235 level, which consist of materials susceptible to moisture pickup from the atmosphere, such as UO_2, etc., will be in sealed containers to prevent a change in moderation while in storage.

(e) In preparing packages for shipment or interim storage prior to shipment, the weighings of special nuclear material will be verified by a production supervisor for assurance against double batching.

b. Arrays by Solid Angle Method - Safe storage arrays may be established by solid angle calculations to determine the safe spacing for interacting units. The solid angle between units will be calculated by the point-to-plane method described on page 34 and illustrated in Figure 25 of TID-7016 (Rev.1). The unit around which the total solid angle is determined will be selected so as to give the greatest spacing within the array, i.e., the most reactive or most central unit. Each unit in such arrays will be spaced from its nearest neighbor by at least twelve inches. The allowable safe interaction for an array will be established using only known or verified multiplication factors applicable to the systems.

Cubic storage arrays to be established by solid angle calculations will be designed and based on the safe neutron interaction criteria as given in Chapter 3, K-1019 (5th Rev.) and summarized as follows:

Storage Unit Control Variable	Mult. Factor, k	Max. All. Interactions of Array (steradians)
Safe cylinder dia.(5" ID max.)	0.58	3.2
Safe mass	0.65	2.5
Safe volume container (max. 4.8 liters)	0.71	1.9
Safe cyl. dia.+(greater than 5")	0.80	1.0

+*As defined in K-1019 (5th Rev.) Section 3.3*

Each unit in the array will be nuclearly safe assuming optimum moderation and full reflector condition. In all cases of solid angle arrays, subcrits will be maintained

*5.9 Cimarron Laboratory - Nuclear Criticality Safety

Control against criticality in the Cimarron Laboratory area is maintained by assurance that only nuclearly safe units are in process at any time and that safe neutron interaction between such units always prevails. The laboratory work-in-process units are conservatively restricted to individual safe U-235 masses and/or safe solution volumes as may be required for sample weighing, preparation and analysis. The individual laboratory process units shall not exceed 350 grams U-235 or 4.8 liters of solution in accordance with the limiting values of Table I, TID-7016 (Rev.1), irrespective of the actual enrichment of the material being analyzed. Further, the entire laboratory work area is administratively restricted to a maximum of six such safe units in process at any one time.

It is the responsibility of the individual laboratory analysts to control the process units which they handle to the nuclearly safe limits and each analyst is specifically instructed regarding the laboratory limits for criticality control. The analysts are considered to be technically qualified and competent to assume the administrative responsibility for maintaining the laboratory control limits. The laboratory Supervisor performs daily surveillance of the laboratory area for compliance to criticality control limits.

The laboratory area is administratively and physically divided to provide six work stations consisting of four laboratory work tables and two work stations in the analytical balance room. Each work station is restricted by the laboratory analysts to a single safe process unit of U-235 (one safe mass unit or one safe volume unit) at any one time.

Safe neutron interaction is assured by the safe spacing between process units provided by the natural arrangement of the laboratory work stations. Demonstration of safe interaction is based on the minimum value of the lattice density method of TID-7016 (Rev.1), Figure 22, which assures the safety of at least ten units in a cubic array with 8 inches minimum open space between units.

The laboratory process units are normally separated by 3 to 5 feet or greater but at all times will be maintained at a separation distance greater than 10 inches including transfers from one work station to another or to and from laboratory storage. All special nuclear material in the laboratory except the work-in-process units will be stored in accordance with the applicable license conditions. The Cimarron Laboratory is an established Material Balance Area for nuclear materials safeguards purposes. The Laboratory Supervisor is custodian of the laboratory material balance area and he controls laboratory storage and authorizes movements in and out of storage.*

*5.10 Typical Calculation - Less than Full Reflection Condition

The following example for nominal reflection is given to typify a calculative method which is used to establish criticality control limits for process units where the conditions of nominal reflection are justified in accordance with the applicable license conditions. Allowance for conditions of less than nominal reflection is not taken to establish limits for the Cimarron activities.

Problem: What is the minimum critical and nuclearly safe infinite cylinder diameter for a nominally reflected system of 20% U-235 solution at optimum moderation?

1. For 20% U-235 enrichment in hydrogen-moderated system:

    a. Unreflected infinite cylinder dimaters = 10.8" (refer Figure 23, TID-7028).

    b. Full water reflected infinite cylinder diameter = 8.2" (refer Figure 23, TID-7028).

2. The minimum critical infinite cylinder diameter is conservatively calculated as the average of the bare and reflected diameters.

$$\frac{10.8 + 8.2}{2} = 9.5\text{" dia. (minimum critical)}$$

3. A safety factor of 93% of the critical dimension is applied to obtain a nuclearly safe cylinder diameter for the system of nominal reflection.

    9.5" x .93 = 8.83"*

*11.0 MANUFACTURE OF NUCLEAR FUEL RODS (PROCESS AREA F)

This section describes the operations for the manufacture of nuclear reactor fuel rods containing uranium oxides at any enrichment. Included are the process and equipment descriptions, equipment layout and the appropriate nuclear safety analyses for the proposed activities.

The equipment layout installed for the fuel rod fabrication activity is given in drawing NPD-107-201 (Rev.1).

11.1 Process Description

Enriched UO_2 powder is converted from UF_6 feed material in the high enriched process equipment (Process Area E-1, refer Section 10). The UO_2 powder is processed to pellets which are used to manufacture nuclear fuel rods consisting of UO_2 pellets encapsulated in metal cladding usually stainless steel.

The UO_2 powder is pelletized and the pellets are sintered at high temperature, sized to finish diameter by centerless grinding, and inspected. In fuel rod manufacture, the UO_2 pellets are assembled into the appropriate length pellet stacks and inserted into metal tubing for outgassing. The pellet stacks are vacuum outgassed in tubing and encapsulated in the finish cladding tubes by welding an end closure. The finished fuel rods are dimensionally inspected and helium leak tested prior to shipment.

11.2 Process Equipment

A plan view of all major process equipment to be used in the fuel rod manufacturing is given on Drawing NPD-107-201 (Rev.1).

The location and description of the fuel rod fabrication process equipment and the related processing of UO_2 powder to fuel pellets which may be used for this activity includes:

1. UO_2 Pellet Production - the following process equipment may be used in the production of UO_2 pellets.

Drawing Item No. In Process Area "B"	Equipment Description/ Operating Stations++
A	Powder preparation hood
B	Pellet press
C	Sintering furnace
In Process Area "A"	
470-101, -102, -103	Pellet presses

405-117, -118	Sintering furnaces
490-101	Centerless grinder
480-101, -102	Inspection work stations

2. Manufacture of Fuel Rods (Process Area "F") - The process equipment to be used in the manufacture of fuel rods includes:

Drawing Item No.	Equipment Description/ Operation Stations++
1	Stainless steel component cleaning
2	First end cap welding
3	Oven for drying cladding components
4	Pellet inspection table
5	Pellet stack assembly - inspection table
6	Work table for loading stacks to glove box
7	Glove box for outgassed stacks of UO_2 pellets
8	Vacuum outgassing furnaces
9	Welding glove boxes
10	Welding machines
11	Helium leak detector
12	Helium mass spectrometer
13	Helium impregnator
14	Fuel rod inspection table
15	Fuel rod packaging station

++ The material inventory at each SNM operating station will be limited to a single nuclearly safe mass of U-235 except for the sintering furnaces which are controlled as safe slabs.

11.3 Process Operating Conditions

The enriched UO_2 powder and pellets will be processed as individually controlled batches of nuclearly safe masses with limited moderation except as otherwise noted herein. The production orders will be conducted by campaigns for the different enrichments with equipment cleanout prior to processing a different enrichment.

The entire process has been analyzed for nuclear criticality safety for normal and abnormal environmental conditions. Accidental criticality during the conduct of this activity is not considered to be credible with the process limits and controls imposed by operating procedures.

The following operational controls are imposed to guard against criticality:

1. Criticality control limits for the nuclearly safe masses are based on conditions of moderation other than optimum. Analytical verification of the moisture content of the UO_2 powder is obtained prior to accumulation of the moderator controlled safe process batches. Prior to this the enriched UO_2 powder is contained in geometrically favorable equipment designed to be safe for full enrichment.

2. In the pellet sintering operation, nuclear criticality control is based on geometrically favorable infinite slab configuration assuming optimum moderation and full water reflection even though moderation of the UO_2 pellets is not credible.

3. In pellet grinding (Area A - item 490-101) where optimum water moderation of the UO_2 grinder sludge might be credible, an inventory control on grinder sludge is imposed for criticality safety. Recorded differences in weights of pellets in and out of the grinder will be used to establish the accumulated safe mass of U-235 in the grinder coolant. The grinder will be cleaned out when the mass limit is realized. Close supervisory control is imposed to guard against double batching.

4. In the end cap welding operation (Area F - Item 9) for fuel rod fabrication, a water cooling coil is used on a chill block in the welding fixture. Since a leak in the cooling coil is considered credible, this operation will be under constant operator attendance while special nuclear material is in a welding glove box and the cooling water is on. In the event a leak occurs the flow of cooling water will be immediately shut off to provide continued moderation control and the leak shall be repaired before operations are continued.

5. Each operation with special nuclear materials will be restricted to a single nuclearly safe mass of U-235 except the pellet sintering operation which is controlled as a geometrically favorable slab configuration.

11.4 Nuclear Criticality Safeguards

The establishment of criticality control limits for process units on the basis of moderation control is justified because the fissile material specifications require the state of moderation to be less than optimum and no single credible mishap will result in a more effective degree of moderation than assumed in the criticality analysis.

1. Nuclear Safety of Process Limits

a. Following are the nuclearly safe batch limits to be used in the UO_2 pelletizing and fuel rod fabrication and the bases therefor. The moderation of the UO_2 will be limited not to exceed an H/X = 2 regardless of the enrichment and controlled by analytical verification of the actual moisture content.

Material	Control	Nuclearly Safe U-235 Mass Limit+
UO_2 Powder	$H/X \leq 2$	20 kg
UO_2 Pellets	$H/X \leq 2$	10 kg

+Reference: TID-7016 (Rev.1), Figure 1.

b. The UO_2 pellet sintering operation will be limited to a safe slab thickness for the enrichment in process and controlled by the number of pellet layers in loading the pressed pellets in the molybdenum furnace boats which are 7" x 7" x 3" high. The nuclearly safe slab thickness for the enrichment in process will be derived by taking 93% of the minimum critical value of Figure 24, TID-7028, assuming full water reflection for the heterogeneous, hydrogen-moderated system. No allowance is needed in this operation to relax the safe slab limits for the less than optimum moderation conditions which actually exist.

c. Accumulations of grinder sludge will be limited to the nuclearly safe masses assuming optimum moderation and full water reflection. The safe U-235 masses for the specific enrichment in process will be taken from Table XIV, K-1019 (5th Rev.).

2. Neutron Interaction

Safe neutron interaction is assured by the controlled mass limits of process units and the spacing between units provided by the natural arrangement of the operating stations. Demonstration of safe interaction is based on the minimum value of the lattice density method of TID-7016 (Rev.1), Figure 22, which assures the safety of at least ten units in a cubic array. The 10 kg U-235 mass limit on process units of pellets contains adequate safety factor to guard against the possibility of double batching. The maximum number of process units in array proximity at any one time will consist of 10 units in the fuel rod fabrication rooms (Area "F"). The minimum distance of any two units in process is 24" edge-to-edge between units in the vacuum outgassing furnaces. Process units in movement will be maintained at a separation distance of at least 10" from other units during transfer from operation to operation.

12.0 PACKAGING OF SNM FOR TRANSPORT

This section presents the detailed demonstrations of nuclear criticality safety in accordance with 10CFR71 for the packaging of special nuclear material for delivery to a carrier for transport.

12.1 Package Models

Safety demonstrations for the following package models used for the transport of special nuclear material are included. License conditions for these package models are given in Appendix A.

Section	Package Model
1.0	Packages of UF_6 Cylinders
2.0	Package Model No. NPD-1-528
3.0	Package Model No. BE-586
4.0	Package Model No. BE-1292
5.0	Package Model No. DOT-6L (Class III shipment)
*6.0	Package Model KM-1*

1.0 Packages of UF_6 Cylinders

Adequate safety demonstrations to show compliance to the requirements of 10CFR71 are given in the following reports:

1.1	Package Model OR-5	-	Refer Report K-1716
1.2	Package Model OR-8	-	Refer Report K-1714
1.3	Package Model OR-12	-	Refer Report K-1714
1.4	Package Model OR-30	-	Refer Report K-1686

Fissile Class II Standards (refer 10CFR71.39)

Undamaged Packages - It is shown above that at least 100 damaged packages in an unreflected array are subcritical. Imposing a safety factor of five and reducing by a factor of four for array reflection shows that at least five undamaged packages (100 ÷ 4 x 5 = 5) are subcritical.

Damaged Packages - A factor of 1.1 is applied for interspersed moderator since the packages are subcritical for conditions of optimum moderation. Imposing a safety factor of two and further reducing by a factor of four for array reflection gives 11 damaged packages subcritical.

For this method of analysis, the allowable number of five undamaged packages is therefore limiting for Class II with each package assigned *ten transport index (50 ÷ 5 = 10)*.

Fissile Class III Standards (refer 10CFR71.40)

Undamaged Packages - It has been shown that at least 100 damaged packages in a bare array are subcritical. Imposing a safety factor of two and reducing by a factor of four for array reflection shows that at least 12 undamaged packages are subcritical.

Damaged Packages - A factor of 1.1 is applied for interspersed moderator since the packages are subcritical for conditions of optimum moderation. A factor of four used for array reflection results in an allowable number of 11 damaged packages as a Class III shipment (100 - 4 x 1.1 = 11.3).

A summary of proposed shipments of BE-586 packages is given below.

Allowable Number of BE-586 Packages

	No. of Packages	*Transport Index/Package*
Class II	5	*10
Class III	11	-

3.3 Procedural Controls

a. Prior to the use of the packaging it shall be ascertained that there are no defects which could significantly reduce the effectiveness of the packaging.

b. Each package will be conspicuously and durably marked with the model number BE-586.

Uranium Oxide Powders:

Enrichment (wt. % U-235)	5.0	4.5	4.0	3.5	3.0
Bare critical mass (kg U-235)	2.77	3.21	3.66	4.00	4.42
Effective mass (kg U-235)	.774	.896	1.021	1.116	1.233
Fraction critical bare	.279	.279	.279	.279	.279
Density exponent	1.442	1.442	1.442	1.442	1.442
Cell volume (liters)	215	215	215	215	215
Effective U-235 array density (g/l)	3.60	4.17	4.75	5.19	5.73
Containment U-235 density (g/l)	40.90	47.34	53.95	58.97	65.15
Critical bare array mass (kg U-235)	92.10	106.6	121.7	133.0	147.19
No. packages critical unreflected	118	118	119	119	119

Fissile Class II Standards (refer 10CFR71.39)

a. Packages of uranium metal

Undamaged Packages - It has been shown above that at least 1104 unreflected packages would be subcritical. Imposing a safety factor of five and reducing by a factor of 13 for array reflection results in an allowable number of 16 undamaged packages (1104 ÷ 13 x 5 = 16).

Damaged Packages - A factor of 1.1 is applied for interspersed moderation since the packages are subcritical for assumed conditions of optimum moderation (refer SC-RR-65-98, pages 679-690). Imposing a safety factor of two and reducing by 13 for reflection, the allowable number of damaged packages is 38 (e.g. 1104 ÷ 13 x 1.1 x 2 = 38).

The allowable number as Class II packages is therefore 16 packages each assigned *3.2 transport index (50 ÷ 16 = 3.2)*.

b. Packages of uranium oxides

Undamaged Packages - The package design prevents water inleakage for normal transport conditions. The uranium oxide products are essentially unmoderated (H/U not exceeding one) under normal conditions of transport. Since the infinite medium multiplication factor (k_∞) is less than one for the undermoderated oxides (refer K-1663, Figure 1, page 3), the number of packages under normal transport conditions is not limited and the allowable number of packages per shipment is determined by consideration of the packages as damaged by the hypothetical accident conditions.

Damaged Packages - The foregoing table shows a minimum of 118 unreflected packages critical. This number is reduced by a factor of four for reflection and a factor of 1.1 for interspersed moderation since the packages are subcritical units for assumed conditions of optimum moderation (refer SC-RR-65-98, pages 679-690). The safe number of 13 packages is thus determined by imposing a safety factor of two for the accident situation (118 ÷ 4 x 1.1 x 2 = 13.4).

In accordance with 10CFR71.39(b), the allowable number of BE-1292 packages containing uranium oxides for Class II shipment is 12 each assigned *3.9 transport index*.

Fissile Class III Standards (refer 10CFR71.40)

The numbers of BE-1292 packages proposed for transport under Class III control are:

a. Packages of uranium metal - Class III Shipments

Undamaged - An undamaged shipment of 42 packages with close water reflection will be subcritical in contact with an identical shipment (1104 - 13 x 2 = 42.5).

Damaged - An array of 33 damaged packages, reflected and moderated, would be subcritical (1104 ÷ 13 x 2.5 = 33.9). We therefore proposed to ship no more than 33 of the BE-1292 packages, containing uranium metal buttons, as a Class III shipment.

b. Packages of uranium oxides - Class III Shipments

Undamaged - The number of undamaged packages is not limiting.

Damaged - An array of 26 damaged packages, reflected and moderated, will be subcritical (118 ÷ 4 x 1.1 = 26.8). We therefore propose to ship no more than 26 of the BE-1292 packages, containing uranium oxides, as a Class III shipment.

A summary of proposed shipments of BE-1292 packages is given below.

ALLOWABLE NUMBER OF BE-1292 PACKAGES

Class II Packages	Max/Packages (kg U-235)	Allowable No. of Packages	*Transport Index/Package*
Uranium metal	14	16	*3.2
Uranium oxides	(see Table I and II)	12	*3.9
Class III Packages			
Uranium metal	14	33	-
Uranium oxides	(see Table I and II)	26	-

*6.0 Package Model No. KM-1

6.1 Package Description

a. Package Category: Fissile Material Only -- Class III

b. Packaging

(1) Model No. KM-1

(2) Construction: The package is a trailer assembly consisting of a fabricated steel slab structure which is bolted to the frame of a specially designed 40-foot trailer. The package is constructed and assembled to form a slab structure (4.5 ± 0.1 inches high x 7 3/4 feet wide x 40 feet long) for containment of the product in transport.

Details of construction are shown on Kerr-McGee Drawings NDA-2-501-17, Sheets 1 through 5, Pellet Shipping Trailer. The drawing details include:

Drawing NDA-2-501-17	Title
Sheet 1	Trailer Assembly
Sheet 2	~~Sections and Details~~
Sheet 3	Sub-Assembly and Details
Sheet 4	Details
Sheet 5	Sub-Assembly and Details

The slab containment structure is fabricated in five sub-assemblies having 3/16" thick high strength steel plate on top, bottom and sides and 1/2" thick high strength steel plate on the ends.

(3) Gross Weight: The total gross weight of the trailer/slab assembly including 15,000 pounds of product is calculated to be 36,000 pounds.

c. Contents - Low enriched (unirradiated) uranium dioxide pellets of enrichments not to exceed 5% U-235 will be transported in the KM-1 package.

(1) Type and form of material - UO_2 fuel pellets will be sealed in plastic sheet, loaded into cardboard boxes and transported in the slab containment assembly assuming conditions of optimum moderation and full water reflection.

(2) Maximum quantity of material per package (i.e., per shipment) will be 15,000 pounds of UO_2.

(3) Maximum radioactivity - A shipment of 15,000 pounds of UO_2 at the maximum enrichment of 5% U-235 is equivalent to about 16.2 curies.

6.2 Package Evaluation

The proposed package satisfies the standards specified in 10CFR71 Subpart C for a Fissile Class III shipment.

Standards of 10CFR71 Subpart C
71.31 General Standards

(a) The packaging materials are such that there will be no significant chemical, galvanic, or other reaction among the packaging components or between the package contents and the packaging.

(b) The package is equipped with positive closures, as shown by the detailed drawings, to prevent inadvertent opening. A positive closure is provided for each of the five sub-assemblies. A security seal which is not readily breakable and which, while intact, will be evidence that the package has not been illicitly opened, shall be attached to each sub-assembly prior to transport.

(c) The package does not have an attached system of lifting or tiedown devices as a structural part of the packaging. The steel slab structure is made integral to the trailer by bolting to the trailer frame.

71.33 Criticality Standards

(a) The containment vessel will be a subcritical infinite slab thickness for the enrichment being transported assuming water inleakage for optimum moderation and fully reflected on all sides by water in accordance with Table I.

TABLE I

UO_2 Pellet Transport

Nuclearly Safe Slab vs. Enrichment

Enrichment (% U-235)	Nuclearly Safe Transport Slab Thickness (Inches)+
≤3.50	4.6
3.51 to 4.30	4.1
4.31 to 5.0	3.7

+ Approximately 96% of maximum safe values taken from or derived by linear interpolation of data in Table VII, DP-1014.

The slab containment vessel constructed in accordance with the drawings will not exceed a 4.60 inch slab thickness. This provides adequate margin of safety against criticality for the shipment of enriched UO_2 pellets not exceeding 3.5% U-235. To maintain a nuclearly safe slab thickness for the transport of enrichments above 3.5% U-235 but not exceeding 5.0% U-235, plywood sheets will be inserted in the containment vessel as needed to reduce the slab to a safe thickness for the enrichment being transported in accordance with Table I. Supervisory inspection and measurement of the slab thickness will be made prior to each shipment to assure compliance.

(b) Liquid Control Leakage - Not Applicable.

71.34 Single Package Evaluation

The proposed Fissile Class III shipments are considered as single package shipments. The ability of the package to withstand normal transport conditions has been evaluated by an engineering assessment. The gross weight of the package, including trailer, slab structure and product was used in the engineering assessment for evaluating the effect of the hypothetical accident condition.

71.35(a) Normal Conditions of Transport

The design and construction of package Model KM-1 provides adequate assurance that under the normal conditions of transport (Appendix A - 10CFR71) there will be no release of radioactive material from the containment vessel and the effectiveness of the package will not be substantially reduced. We contend that the safety of the proposed Class III shipments as reasonably demonstrated herein for the hypothetical accident conditions obviates the need for redundant detailed considerations that the package will withstand the normal conditions of transport without damage affecting safety.

71.35(b) Single Package Standards - Normal Transport Conditions

The package will be subcritical as demonstrated under Section 71.33 above. The inner packaging is designed to prevent any damage to the UO_2 fuel pellets under normal transport conditions. The side closures to the containment vessel have a 5/16" thick neoprene sponge to prevent inleakage of water under normal transport conditions. Nevertheless, the containment vessel will be a subcritical infinite slab thickness assuming optimum moderation.

71.36 Hypothetical Accident Conditions - Single Package Standards

An engineering assessment (see Section 6.5) demonstrates that the steel containment structure will maintain the subcritical slab configuration in the event the shipment were subjected to and damaged by the hypothetical accident conditions. No radioactive material would be released from the package and the package will be subcritical assuming optimum moderation with close water reflection on all sides of the damaged package.

71.37 Array of Packages - Not applicable. The proposed Fissile Class III shipments are single package shipments.

71.38 Fissile Class I Standards - Not applicable.

71.39 Fissile Class II Standards - Not applicable.

71.40 Fissile Class III Standards

(a) The product containment vessel is a subcritical slab thickness for infinite dimensions with the shipment closely reflected on all sides by water (refer Section 71.33). The size of the slab in the Model KM-1 package is such that an identical shipment in contact with it in a normal transport situation would not significantly increase the reactivity.

(b) The criticality standards of Section 71.33 and the engineering assessment of Section 6.5 demonstrate that the shipment would be subcritical if subjected to the hypothetical accident conditions with resultant optimum moderation of the package contents and close reflection by water on all sides of the package.

6.3 Procedural Controls

a. We give assurance that adequate inspections and assessments will be made by a qualified engineer to verify that the KM-1 packages are fabricated in accordance with the approved design. Prior to the first use of the packaging, it shall be ascertained that there are no defects which could significantly reduce the effectiveness of the packaging. The thickness of the slab on which reliance for criticality safety is based will be checked by a "no-go" gage technique to assure compliance.

b. Each package will be conspicuously and durably marked with the model number KM-1, a serial number and the Department of Transportation Special Permit Number.

c. Supervisory control will be used for packing and inspection for proper assembly, loading, closure and sealing of containment vessels. Operating procedures will be established and used to assure that the determinations, controls and safeguards imposed on the material and package for safe transport are adhered to, including instructions regarding:

(1) Proper loading, marking and labeling of product inner containers and the package assembly.

(2) Proper assembly, closure and inspection of sealed containment vessels (gaskets will be inspected for integrity prior to use and replaced if damaged to assure leaktightness for protection of product quality).

(3) Inspection of the package assembly for damage after each use.

(4) Maintenance of shipping records; and

(5) Issuance of instructions to the driver of the carrier of special controls and precautions and the notification action to be exercised in the event of accident or delay.

6.4 Transport Method

Package Model KM-1 will be transported only as Fissile Class III shipments by private or common motor vehicle in accordance with the applicable U.S. Department of Transportation approved controls.

6.5 Engineering Assessment of Structural Integrity for Fissile Class III Package - Model KM-1

6.5.1 Introduction

The USAEC Rules and Regulations - Title 10 - Atomic Energy Part 71, "Packaging of Radioactive Material for Transport", specifies the requirements that must be met by a shipping container for radioactive materials.

The following engineering assessment has been made to demonstrate that the model KM-1 shipping trailer will meet all of these requirements. Calculations are included for the following structural considerations:

a) Maintain structural integrity after a thirty (30) foot fall or at a 30 mph head-on collision.

b) Maintain structural integrity at 1475° for thirty (30) minutes.

c) Failure to puncture from the impact of a 40-inch fall on a six (6) inch diameter blunt post.

A water immersion test to prove that the container is watertight is not required in this design.

6.5.2 Analytical Concepts

When an object is subjected to an impact load, considerable energy can be absorbed by the object. The force of the impact load therefore becomes a function of the energy absorbed. The impact force P = (x)w where x is the multiple of g-loads and w is the normal weight. Several handbooks (Mechanical Engineers, Baumeister and Marks, among them) say x = f (h,e) where he is the height of fall and e is the energy absorbed by the object on impact. The first variable of this function is easily obtainable; i.e., h, by AEC specification. The energy absorbed by the object is a different matter and is a function of several additional variables.

Baumeister and Marks suggest that in the case of a beam, the energy absorbed can be safely assumed to be in direct relation to the maximum deflection of the beam. They suggest that the following can be used as a conservative estimate of the "g-load" of a fall from a given height.

$$P = W\left(1 + \sqrt{1 + \frac{2h}{y}}\right) \qquad \text{(Equation 1)}$$

Where h - height of fall; and y - maximum deflection of beam.

In order to examine the actual problem in detail and to determine finite values, the trailer and slab construction are considered a composite beam. (A moment of Inertia and sectional modules has been calculated for this beam.) The following force diagram indicates the worst possible load the theoretical accident may produce on the trailer.

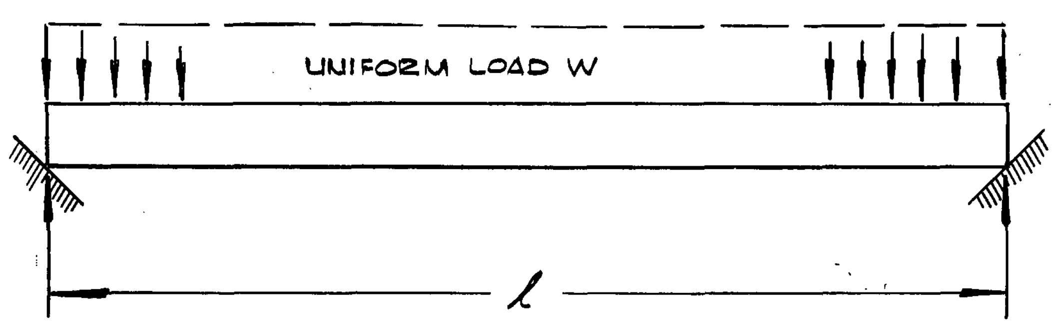


FIG.#1

Considering the trailer/slab as a beam, the maximum deflection is a function of the maximum allowable stress in the outermost fiber of the beam section.

This maximum stress is chosen as the yield strength of the material used in fabrication and a maximum force is determined. This force will then be applied to the beam to determine the maximum deflection. This maximum deflection is assumed to occur at impact. All of these values are within the elastic state of the material (yield strength or less), and, therefore, the classical beam equations can be safely applied.

In Figure 1 the maximum force is

$$W = \frac{8\ Z\ Sm}{l} \qquad \text{(Equation 2)}$$

$$\therefore\ Ym = \frac{5}{384} \left(\frac{8\ Z\ Sm}{l}\right) \frac{l^3}{EI_T} \qquad \text{(Equation 3)}$$

I_T = Moment of Inertia of the Trailer/Slab

As can be seen the beam loading in Figure 1 indicates a uniform loading which exists in the trailer/slab. The product load is evenly distributed as well as the structure of the trailer/slab.

It should be noted that additional conservatism has been introduced by keeping all stresses within the elastic limit. The additional energy absorption of partial yield or plastic deformation is not considered in the analysis.

Where plastic deformation is considered as part of any of the following analysis, its contributions are calculated.

The conservatism of keeping all stresses in the elastic limit during a catastrophic accident has been incorporated in this analysis in order to prove more readily by analytical means that the proposed design meets the function as well as the intent of the AEC regulation. Because of the conservative assumptions, a reduction of allowable yield stress by a safety factor has not been made. It is felt that a safety factor is built in by the fact that permanent deformation is energy absorbed that reduces the impact load very rapidly.

Several additional loading configurations other than the above are analyzed in the following text. Similar conservatism is practiced in setting up the problem for analysis. The above loading is, however, the most severe for this trailer/slab configuration.

6.5.3 Analysis

a. Weight Calculations

Note: All weights, moments and section moduli are taken from the AISC Steel Instruction Manual where applicable; all others have been calculated. The trailer design drawings identify the following list of materials.

(1)	4 x 1 5/8 x 7.25 #/ft]	240' used	1,740#
	$I = 4.5\ in^4$; $Z = 2.3\ in^3$;		
	$A = 2.12\ in^2$; $y = 2$ in.		
(2)	1 1/2 x 2 x 3/16 x 2.12 #/ft L	120' used	254#
(3)	3/16 plate 7.65 $\#/ft^2$	$320'^2$ used	2,450#
(4)	3/16 plate 7.65 $\#/ft^2$	$320'^2$ used	2,450#
(5)	1 1/2 x 2 x 3/16 x 2.12 #/ft	120' used	254#
(6)	4 x 1 5/8 x 7.25 #/ft]	240' used	1,740#
	$I = 4.5\ in^4$; $Z = 2.3\ in^3$;		
	$A = 2.12\ in^2$; $y = 2''$		
(7)	2 x 3 x 1/4 x 4.1 #/ft	80' used	328#
	$I = .39 in^4$; $Z = .26\ in^3$;		
	$A = 1.19\ in^2$; $x = .49$ in		
(8)	8 x 1 7/8 x 8.5 #/ft]	80' used	680#
	$I = 23.7\ in^4$; $Z = 5.9\ in^3$;		
	$A = 2.49\ in^2$; $y = 4$		
(9)	14 x 9 1/3 I = Beam (fab) 62.5 #/ft	80' used	680#
	$I = 813.4\ in^4$; $Z = 116.3\ in^3$;		
	$A = 18.5\ in^2$; $y = 7$		
(10)	3/16 Plate Cross Members 6 Places		
	7.65 $\#/ft^2$ 24 ft^2		184#
(11)	3/16 Plate Cross Members 10 Places		
	7.65 $\#/ft^2$ 68 ft^2		520#

(12) Running Gear 4,500#

(13) Miscellaneous Hardware Bolts, Closures, Latches, etc. 900#

TOTAL - 21,000#

(14) The maximum product gross weight will be 15,000# of product carried on the trailer/slab. This gives a total gross weight of the trailer/slab, with product included, of 36,000#. This weight is used as the trailer/slab weight in all calculations.

b. Section Calculations

The section sketched on Sheet Two of the trailer design drawings represents the critically stressed section. It has been calculated that the moment of inertia, section modulus, area and extreme fiber distance are $I = 6{,}021\ in^4$, $Z = 450\ in^3$, $A = 99.18\ in^2$ and $y = 13.4\ in^1$ (ave) respectively.

c. Impact/Fold-Over Analysis

The extreme fiber material and the plate material will be made from ASTM 514 or ASTM 517 material. This material has a minimum yield of 100,000 psi and ultimate of 125,000 psi.

The maximum yield is calculated from Equation 3, Section 6.5.2.

$$Ym = \frac{5}{384}\left(\frac{8\ Z\ Sm}{1}\right)\frac{1^3}{EI_T}$$

$Ym = \cancel{4.75}$ 6.0 inches

Where:
$E = 30 \times 10^6$ psi
$I = 6{,}021\ in^4$
$Z = 450\ in^3$
$Sm = 100{,}000$ psi
$1 = 480"$ or $40'$

The load to create this maximum deflection is

$$W = \frac{8\ z\ Sm}{1}$$

$W = 750{,}000\#$ ✓

$Sm = 100{,}000$ psi
$Z = 450\ in^3$
$1 = 480$ in

P from Equation 1, Section 6.5.2, is

$$P = W\left(1 + \sqrt{1 + \frac{2h}{y}}\right)$$

$P = 480{,}000\#$

$y = 4.75$
$h = 360$ in. or 30 ft.
$W = 36{,}000\#$

The comparison between the 750,000# maximum load and the 480,000# estimated impact load indicate that the elastic limit will not be exceeded on impact.

d. Impact Energy Absorption

The energy absorption from impact cannot always be analyzed as above. The end loading will therefore be checked from an energy balance standpoint suggested in "Design of Weldments" by O.W. Blodgett. This theory indicates that the energy that can be absorbed by a structure is a function of its material volume and its modulus of resilience. It is further indicated that the impact energy and the energy absorption capacity can be equated to establish the maximum impact energy without failure.

From "Design of Weldment", the modulus of Resilience of ASTM 514 steel is given as U = 200 in - #/in^3.

The impact energy is equal to $1/2 \frac{W}{g} v^2$ and the material volume is equal to

$$\frac{\text{Steel Weight in \#}}{\frac{\text{cu in}}{\#}\text{ of Steel}} \quad \text{or} \quad \frac{21{,}000 - 4{,}500}{.281}$$

$$V_m = \frac{16{,}500}{.281} = 58{,}800 \text{ in}^3$$

We can then compare the two energy values; the first; how much can be absorbed:

$$E_a = V_m U = 11.76 \times 10^6 \text{ in. - lb.} \qquad \text{ASTM 514}$$

= 58,800 x 200

and the impact energy which is:

$$E_I = 1/2 \frac{W}{G} V^2 = \cancel{6.4 \times 10^6} \text{ in - lb.}$$ X 12.9 x 10^6 in - lb.

It is further stated that the ultimate toughness for ASTM 514 steel is an estimated 9,400 in - lb/in^3. This calculates an ultimate E_a as

$$\text{Ultimate } E_a = V_m U_u = 9{,}400$$

$$= 552 \times 10^6 \text{ in - lb.}$$

$\frac{11.76 \times 10^6}{6.4 \times 10^6}$ c 1.8

The total impact energy of 6.4 x 10^6 in - lb. is less than the elastic limit energy absorption of 11.76 x 10^6 in - lb., by a factor of 1.8, and the ultimate energy absorption capacity is 86 times greater than the impact energy.

The 15,000# product load component of the W/g factor used in the impact energy equation will absorb considerably more energy than allowed for in these calculations. In consideration of the above, the yield strength of the material will not be exceeded in a thirty foot drop, and the ultimate strength will not be approached within a factor of 86. This indicates that the results above using the beam deflection as a measure of energy absorption is a reasonable assumption since the numerical results are approximately the same.

e. Vertical or Corner Impact

Assuming an impact from the 30 foot fall or a head-on 30 mph collision, a force diagram as indicated in Figure #2 below would apply.

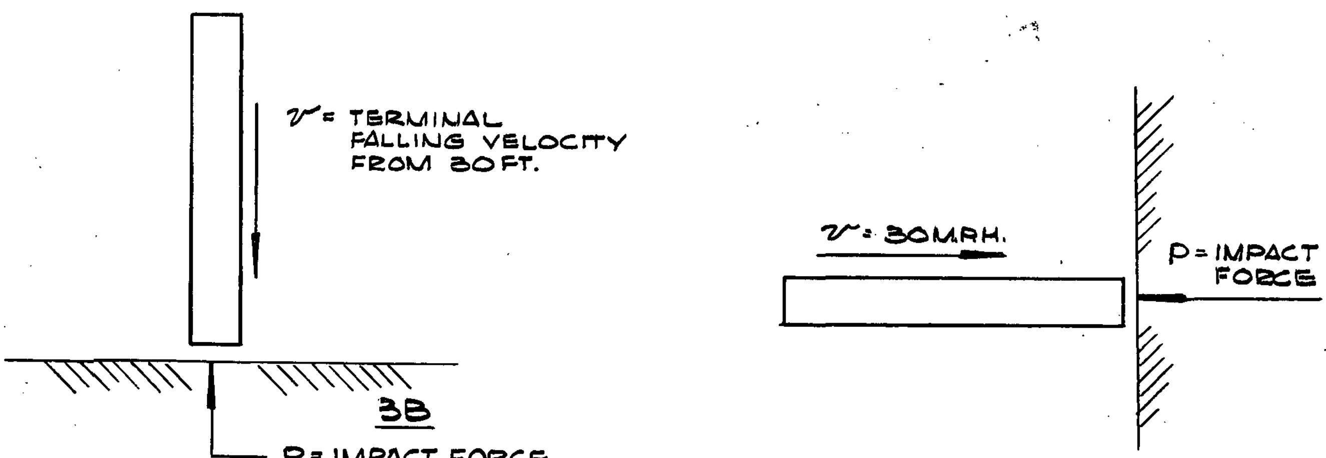


FIG. #2

In Figure 3.B the force P is equal to the normal weight times the impact g-load factor (x). This factor (x) is a function of the energy absorption as described heretofore and related to the strain induced in the material.

The buckling failure limit for this beam will also be checked against the applied impact load P. It is assumed that if a static buckling force (f) is at least three to four times higher than the impact load, then the beam will probably not fail at impact due to buckling.

P can be expressed in the following equation (a variation of Equation 1 in Section 6.5.2).

$P = W\left(1 + \sqrt{1 + \frac{2h}{e}}\right)$ Where e is the strain involved due to the impact load and h = 360 in.

In order to determine e the following equation may be used: P/A = stress in member and this stress should not exceed the yield strength.

A = 99.18 Sy = 100,000 psi

$P = 9{,}918{,}000\# = 9.92 \times 10^6 \#$

Per Machinery's Handbook, J.B. Johnson's formula for critical buckling is

$P_{CR} = A\, Sy\left(1 - \frac{Q}{4r^2}\right)$ Where $Q = \frac{Sy l^2}{\pi^2 E}$

Sy = 100,000 psi
l = 480 in.
$E = 30 \times 10^6$
$Q = 78 \text{ in}^2$
r = 11 in.

$P_{CR} = 99.18\,(10^5)\left(1 - \frac{78}{4r^2}\right) = 99.18 \times 10^5\,(1 - .161)$ ✓

$= 8.4 \times 10^6 \#$

Therefore, the critical buckling load is $8.4 \times 10^6 \#$ or less than the critical compressive load.

Using the formula $\frac{Pl}{AE} = Ym$ we have $E = \frac{P/A}{Ym/l}$

Ym = 1.33 inches; where $P = 8.4 \times 10^6$, $A = 99.18 \text{ in}^2$ and $E = 30 \times 10^6$ psi.

Returning to the impact load formula above, we have:

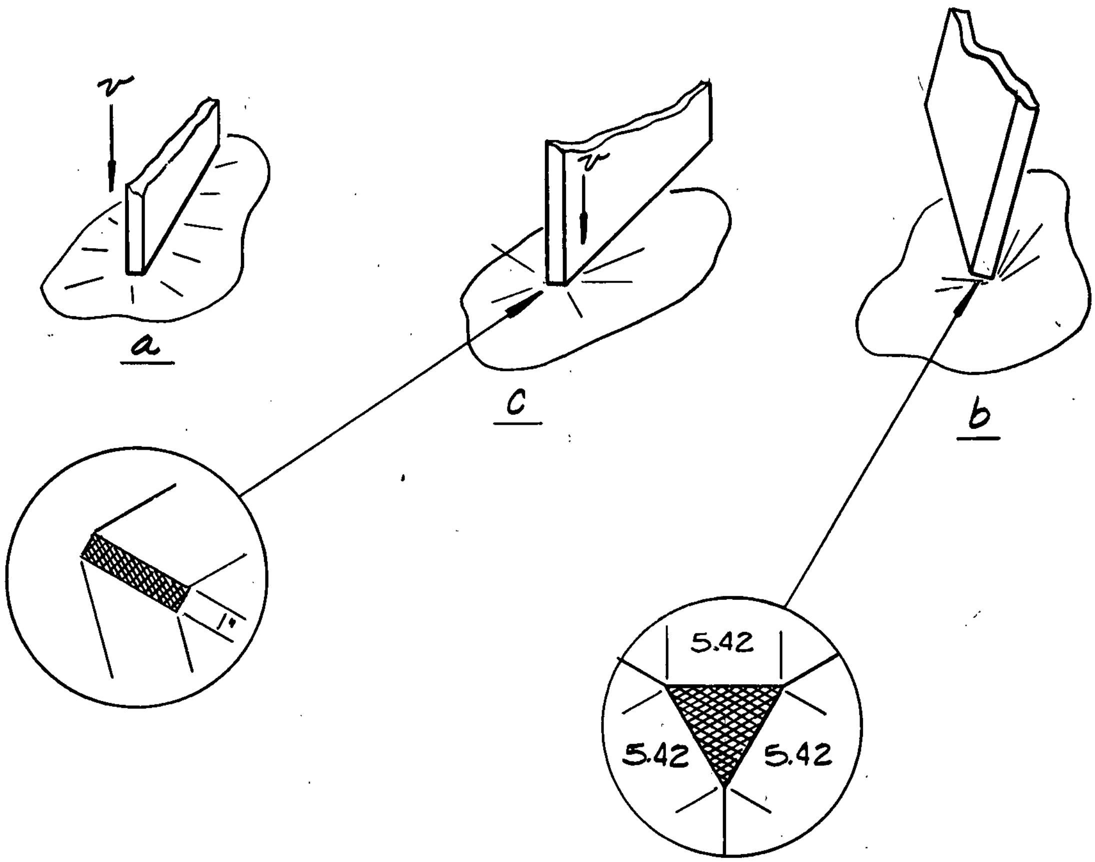
$P = w\left(1 + \sqrt{1 + \frac{2h}{e}}\right)$

$= 36{,}000\left(1 + \sqrt{1 + \frac{720}{1.33}}\right) = 36{,}000\,(24.3)$

$= 872{,}000\# = .872 \times 10^6$

This impact load is 1/8 as high as the critical buckling load. The impact load resonance factor as mentioned in Marks & Baumeister where the impact surface is considered rigid was considered in the above calculation. Therefore, it is safe to assume the beam will safely survive the fall in this position.

What will happen locally on the end that strikes the rigid surface is examined below. (Figure #3)



96
20

FIG. # 3

We have calculated an impact load of .872 x 10^6# above. The surface area if it strikes square with the end is 1,920 in^2 of steel covered surface. This surface transmits the load to 99.18 in^2 of column. The force per square inch on the end surface is

$$F = \frac{.872 \times 10^6}{.1920 \times 10^4} = 455 \text{ psi}$$

This is a relatively small unit force and would not be expected to cause damage.

Blodgett - ay

However, it is unexpected that a fall would occur where a flat, rigid surface or a square hit would be the case. A corner hit would be the more severe. Let us assume that a load of 75,000 pounds per square inch (Lm) is a maximum allowed. Then an area of

$$\frac{P}{L_m} = A = \frac{\overset{.872}{\cancel{.946}} \times 10^6}{.75 \times 10^5} = 12.6 \text{ in}^2$$

$P \quad \frac{\#}{\#/in^2} = in^2$

would be required to support the impact at this allowable load. Cutting off the corner or brinelling it to a flat that is equal to 12.6 in^2 would result in a flattened area about one inch wide (Figure 3.C) on the edge or a corner flattened to 5.42 inches (Figure 3.B) on a side. Neither of these cases would cause the slab holding the product to be changed.

f. Additional Impact Considerations

$\frac{1}{2} \times 5.4^2$

(1) Product (Load) Characteristics

It should be noted that the inner containers of pellets within the slab are designed for maximum energy absorption. The 15,000 pounds of product therefore will not contribute to the impact loads as a solid attached mass of the same weight would. This represents approximately 40% of the total mass involved. There has been no attempt to include these package characteristics in the impact calculations but in every case it would tend to reduce the numbers.

(2) Product Compartment Integrity

The product load is free to shift and will cause a secondary impact when the trailer/slab experiences a sudden stop. Each compartment contains 3,000# of product. The package is extremely energy absorbent ? and this factor will greatly reduce the g-loads experienced by its impact.

Regardless of this fact, let us assume the package contents meet the sides of the compartment at the maximum g-load calculated above. This would give a load of L = 3,000 (24.3) = 72,900#. The area of impact is 4.5 (96) = 432 in^2. This gives a surface load of 168.8 psi. It would give a shear load in a 3/16 inch thick plate of 4.050 psi which is well below the shear strength of the material (75,000 psi). ?

6.5.4 Fire Damage

The accident theorized, includes a fire that heats the trailer/slab to 1475°F. The question arises as to whether or not the trailer slab will support its weight and product load at this temperature.

The yield strength at 1500°F has been established for carbon steel as 8,000 psi min.

The stress in the trailer/slab beam in a static condition may be expressed as

$$S = \frac{wl}{8Z} \quad \text{Uniformly Loaded Beam (See Equation 2, Section 6.5.2)}$$

$= 3{,}060$ psi

$w = 36{,}000\#$
$l = 480$ in
$Z = 450\ in^2$

This is well below the minimum yield strength of 8,000 psi; therefore, no failure or movement in the structure will occur at 1475°F.

6.5.5 Puncture Analysis

The trailer/slab product compartments must withstand a puncture by a six inch diameter blunt instrument after a drop of 40 inches.

It is a conservative assumption that the puncture occur entirely in shear.

Shear Area $A_s = \pi dt = 3.14 (6)\ 3/16 = 3.50\ in^2$

Max. Shear Stress $S_s = .75\ (100{,}000) = 75{,}000$ psi

Total Load to Shear $P_s = S_s A_s = 264{,}000\#$

The impact load from a 40 inch fall is

$$P = W\left(1 + \sqrt{1 + \frac{2h}{y}}\right)$$

$W = 36{,}000\#$
$y = 4.75''$
$h = 40''$

$P = 5.1\ W = 184{,}000\#$

$P_s/P = 1.43$

This is a safety factor of 40%.

6.5.6 Slab Cover and Closure Latch

The slab cover will be pivoted at the bottom on a continuous heavy duty piano hinge and latched with nine studs as shown in the drawing. Each group of three studs will be wedged shut by a single plate.

The impact load against the plate could be as high as 72,000# (see above). The latching wedge will be 3/16 thick material.

The studs will be 1/2 inch diameter with 3/4 inch diameter heads.

Therefore the shear area will be A = πd per stud; or A = 9 πd 3/16 for the cover. Where d = 3/4 inch, we have 3.77 in^2 of shear area. With an impact load of 72,000 divided by the area, we have a stress of 19,000 psi or one-fourth of the shear strength. The tensile stress on the stud is equal to Sm = P/A where P = 72,000 and A = 9 $(d/2)^2$ (d = 1/2 inch) or 30,800 psi or about 1/3 of the minimum yield of 100,000 psi.

6.5.7 Conclusions

A trailer/slab (package model KM-1) fabricated in accordance with the attached drawings will withstand all accident criteria outlined in the AEC regulation, without deformation of the slab.

It is assumed that this analysis has proven, with adequate conservatism, that the criteria will be met and that proof testing is unnecessary. Normal nondestructive inspection shall assure that material and dimensional specifications have been met.

6.5.8 Attachments

Kerr-McGee Drawings - NDA-2-501-17 (Sheets 1 through 5).

APPENDIX A

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Appendix A
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License Conditions

for the

Nuclear Division Cimarron Facility

Set forth herein are the technical and administrative specification within which the activities involving special nuclear material are conducted under authority of Special Nuclear Material License No. SNM-928, issued to Kerr-McGee Corporation.

1.0 Authorized Places of Use

Nuclear Division Cimarron Facility (major site)
Kerr-McGee Research Center (off-site minor uses)

1.1 Possession Limits

The following quantites of special nuclear material are authorized:

(a) Nuclear Division Cimarron Facility: 3000 kilograms U-235
(b) Kerr-McGee Research Center: 500 grams U-235

1.2 Authorized Activities

The following types of activites are authorized:

1.2.1 Nuclear Division Cimarron Facility -

1. Manufacture of enriched uranium nuclear fuels, including:
   a) Conversion of enriched uranium hexafluoride to various other uranium compounds.
   b) Pelletizing of enriched uranium compounds *and manufacture of nuclear fuel rods*.
   c) Reduction of enriched uranium tetrafluoride to uranium.
   d) Manufacture of enriched uranium alloys and intermetallic compounds.

2. Recovery of uranium values from unirradiated scrap materials containing enriched uranium.

3. Laboratory operations including analytical and quality control services in the areas of chemical, metallurgical and physical testing and measuring of enriched uranium materials.

1.2.2 Kerr-McGee Research Center -

Analytical services in the areas of emission spectrography, x-ray diffraction, atomic adsorption, and other special laboratory tests.

2.0 Organization, Administration and Personnel

2.1 Organization Structure

The Cimarron Facility is operated by the Nuclear Division Manufacturing Department. Responsibility for the control of all activities, personnel, equipment and materials at Cimarron Facility resides with the Facility Manager and the Manufacturing Department. The Cimarron Facility Manager reports to the Manufacturing Manager, Nuclear Division, who reports in turn to the Vice President, Nuclear Division, who is responsible to the President of the Kerr-McGee Corporation.

2.2 Functional Organizations for Health-Safety

Because of public law and the Company's concern for the well-being and protection of its employees and the general public, the Nuclear Division will maintain an effective Radiological Protection Program and a Nuclear Criticality Safety Program.

To conduct the Health-Safety programs, organization components are established and defined in the Nuclear Division to provide adequate facility management, division and facility level audits of program effectiveness and to reasonably minimize the reporting lines for maximum effectiveness in functional control.

The following functional organizations will be in effect in the Nuclear Division.

2.3 *Radiological Protection Functional Organization

The radiological protection organization is responsible to establish and maintain the program elements designed to ensure the protection against radiation of employees at the operating facilities of the Nuclear Division and of the community.

2.3.1 The radiological protection organization is defined as that component of the Nuclear Division with designated responsibility to:

a) Establish and approve the program criteria and standards for radiological protection;
b) Provide authoritative professional advice and counsel to Division and Facility management on matters of radiation protection;
c) Conduct and monitor the authorized activities in such manner to protect the employees and the general public; and
d) Measure the effectiveness of the radiological protection program.

2.3.2 Assigned responsibilities for the radiological protection program and the minimum qualifications of personnel shall be:

1. The Vice President, Nuclear Division, is responsible for all Nuclear Division activities, including development, marketing, engineering and manufacturing. He specifically approves and modifies the criteria and standards of the radiological protection program and organization. He shall hold a science or engineering degree, have at least five years in a management position, at least three years of which shall have been in a nuclear activity which would develop an understanding of radiation problems, controls for radiological protection and nuclear criticality safety.

2. The Radiological Science Officer will report functionally to the Division Head on radiological protection matters. He is responsible to prepare, evaluate and recommend the program criteria and standards and to audit the radiological protection program. The Radiological Science Officer shall be professionally qualified with a bachelor degree in science or engineering with five years experience in assignments involving radiation protection. He shall be capable of providing authoritative professional advice and counsel on matters of radiation protection.

3. The Licensing and Regulation Officer is responsible for coordinating license matters, reviewing operating procedures and criteria and standards of the health-safety programs. He is also responsible to audit the health-safety programs for license and other regulatory compliance. He reports functionally for radiological protection program elements to the Vice President, Nuclear Division, and will serve as liaison for the Nuclear Division with external regulatory agencies. The Licensing and Regulation Officer shall hold a physical science or engineering degree with at least three years experience relating to safety in a nuclear field involving special nuclear materials and one year experience in criticality safeguards. He shall have demonstrated a proficiency in understanding the regulations for nuclear activities and be capable of performing criticality calculations relating to the Nuclear Division activities. He shall be capable of providing authoritative professional advice and counsel on regulatory matters, radiation protection and criticality control.

4. The Cimarron Facility Manager is responsible for the control and safe conduct of all activities at the Nuclear Division Cimarron Facility. Operating procedures, which specify operating steps within the approved radiological safety criteria and standards, shall be written and have the prior approval of the Facility Manager, who functionally reports to the Division Head on matters of radiological protection.

The Cimarron Facility Manager shall hold a bachelor degree in science or engineering with five years experience at a nuclear facility with activities involving special nuclear materials and at least three years experience in a supervisory position. He shall have demonstrated a proficiency to manage the Cimarron Health-Safety Programs and to identify process changes which require radiological or criticality analysis.

5. The Production Section, Cimarron Facility, conducts the production activities involving special nuclear materials. Personnel health and safety is considered an integral part of line supervisory responsibilities.

   a. The Production Superintendent is responsible for the safe conduct of the production activities in accordance with approved operating procedures. He reports to the Cimarron Facility Manager. He shall have a bachelor degree in a technical field with two years supervisory experience in a nuclear facility or ten years experience in working with special nuclear materials. The Production Supervisor shall be knowledgeable of the approved criteria and standards for radiological protection and nuclear criticality safety.

   b. Shift Supervisors report to the Production Superintendent and it is their responsibility to assure that the operating procedures are followed in the performance of the production activities. Shift Supervisors shall have a bachelor degree with two years experience in working with special nuclear materials or a high school diploma with five years experience in special nuclear material processing. The Shift Supervisors shall be thoroughly familiar with the Cimarron production activities and be knowledgeable of the approved operating procedures.

6. The Cimarron Health-Safety Section is responsible to conduct the radiation protection program at the facility and perform surveillance of the facility activities for compliance to approved design and operating procedures. The Section functionally reports on radiological protection matters to the Cimarron Facility Manager and is administratively independent of process supervision of the facility production activities. The Section shall consist of at least one qualified and experienced professional health-physicist in a full time health and safety capacity. The Section shall conduct and evaluate the environmental monitoring program, a bioassay program, and employee training in the radiological protection program aspects. The Section shall maintain all radiation exposure records required by regulatory agencies.

A Health-Physicist shall have a bachelor degree in science or engineering with three years experience in radiation monitoring, personnel exposure evaluations, with one year experience in auditing or inspecting nuclear facilities. He shall be familiar with the facility production activities and operating procedures and be knowledgeable of radiation protection regulations.*

2.4*Nuclear Criticality Safety Functional Organization

The nuclear criticality safety organization is responsible to establish and maintain the program elements designed to control against an accidental criticality and to measure the effectiveness of the criticality control program.

2.4.1 The nuclear criticality safety organization is defined as that component of the Nuclear Division with designated responsibility to:

a) Establish and approve the criteria and standards for criticality control;

b) Provide authoritative professional advice and counsel to Division and Facility Management on matters of nuclear criticality safety;

c) Conduct and monitor the authorized activities in such manner to guard against the occurrence of an accidental criticality; and

d) Measure the effectiveness of the nuclear criticality safety program.

2.4.2 Assigned responsibilities for the nuclear criticality safety program and the minimum qualifications of personnel shall be:

1. The Vice President, Nuclear Division, is responsible for all Nuclear Division activities and he specifically approves and modifies the criteria and standards of the nuclear criticality safety program and organization. For minimum qualifications, see Section 2.3.2-1.

2. The Nuclear Safety Officer will report functionally to the Division Head on matters of nuclear criticality safety. He is responsible to prepare, evaluate, and recommend the program criteria and standards and to audit the nuclear criticality safety controls.

The Nuclear Safety Officer shall be professionally qualified and experienced in the area of nuclear physics or nuclear engineering. He shall hold a physical science or engineering degree with at least three years experience in a nuclear field and one year experience in criticality safeguards work. The Nuclear Safety Officer shall have a demonstrated proficiency in carrying out criticality calculations and applying them to the types of operations conducted at the Cimarron Facility.

3. The Licensing and Regulation Officer is responsible for coordinating license matters, reviewing the program criteria and standards and facility operating procedures, verifying and approving criticality calculations and control limits and, auditing for license compliance and to measure the effectiveness of the Nuclear Criticality Safety Program. He functionally reports to the Division Head on nuclear criticality safety matters. For minimum qualifications, see Section 2.3.2-3.

4. The Cimarron Facility Manager is responsible for the control and safe conduct of all activities at the Nuclear Division Cimarron Facility and he specifically reviews the criteria and standards of the nuclear criticality safety program, and approves facility operating procedures which specify operating steps within the approved program standards and criteria. He reports functionally to the Division Head on matters of criticality control. For minimum qualifications see Section 2.3.2-4.

5. The Production Superintendent is responsible for the safe conduct of the production activities in accordance with approved operating procedures. For minimum qualifications, see Section 2.3.2-5a.

6. Shift Supervisors are responsible to the Production Superintendent for adherence to the approved operating procedures in the conduct of production activities. For minimum qualifications, see Section 2.3.2-5b.

7. The Cimarron Health-Physicist is responsible for surveillance of the facility activities for compliance to approved design and operating procedures for nuclear criticality safety. For minimum qualifications, see Section 2.3.2-6.*

3.0 Radiological Protection Program Specifications

In addition to the aforementioned requirements of organizational capability and administrative controls, the following program elements are provided at the Cimarron Facility to protect health and minimize danger to life and property.

*General Program Elements

The radiological protection program for the Cimarron Facility is established and maintained to ensure the protection against radiation of employees and residents of the community. The Nuclear Division operating philosophy is to keep personnel exposures to radiation at the lowest practicable level. The radiological protection program shall include as a minimum: the evaluation of release of radioactive effluents and materials from the Cimarron site; establishment of procedures to control contamination, exposure to individuals, disposal of wastes and integrity and reliability of radiation detection instruments including appropriate reports and records associated therewith.*

3.1 Facility Design

Consideration of potential health problems in the environs of the Cimarron Facility played an important part in the facility design. The facility is designed, built, equipped and maintained to satisfactorily insure the radiological, fire and structural safety for the radioactive material processes. The facility design and contamination control programs provide for the occupational safety of employees and simultaneously prevent undue contamination of the surrounding area. Modifications to the facility shall consider the radiological aspects of the proposed activity.

3.2 Equipment

To assure a high degree of efficiency and effectiveness for the radiological protection program and its control systems, the following equipment is provided and used at the Cimarron Facility.

3.2.1 Process Equipment

The process equipment, gloveboxes, hoods, enclosures and instrumentation shall be of proper safety engineering design to insure a high degree of containment, personnel protection, and material accountability.

3.2.2 *Ventilation and Exhaust Systems

The plant is designed to be slightly pressurized at all times with the general plant air primarily discharging through roof vents. The exhaust systems for process equipment and operating areas are designed for effective control of airborne contaminants generated in processing. Special blowers, absolute filters and exhaust ducts are utilized in areas of high airborne contamination potential. Adequacy of ventilation control within the facility will be determined by air monitoring.

Ventilation and exhaust systems are provided to meet the airborne concentration limits of 10CFR20.103 without an allowance for personal respiratory protection during routine operations. Airborne contamination will be minimized and effectively controlled by the following criteria.

1. Ventilation equipment shall provide that the direction of air flows be from areas of lesser contamination to areas of higher contamination. The actual and potential conditions shall be considered in the design of the ventilation equipment.

2. Effluent air velocities and duct design shall minimize plateout aggregation.

3. Containment of airborne contamination shall be done wherever practical by ventilation hoods and gloveboxes, plant exchange, and exhausting systems. The containment enclosures shall maintain the breathing zones in operational areas at air concentration levels of radioactivity at or below the limits of 10CFR20 under normal and foreseeable abnormal operating conditions as determined by air sampling. Ventilation hoods and gloveboxes will be constructed predominantly of metal for fire resistance, with glass or fire resistant plastic windows.

4. Containment hoods shall be designed to provide suitable air flow for containment and open face laboratory-type hoods will have a hood face velocity of at least 100 lineal feet per minute during operations. The air flow at hoods shall be measured at installation and monthly during normal operation. Any reduced face velocity which approaches the minimum effective capture velocity will be corrected. When containment of radioactive dusts by conventional hoods is not feasible, gloveboxes will be used.

5. Exhaust systems for potentially contaminated airborne effluents shall be equipped with absolute filters. Filters shall be 99.97% efficient for removal of 0.3 micron particulate matter. The differential pressure on at least one filter in each effluent air streem shall be measured monthly. A filter shall be replaced at a pressure differential in excess of 4" of water or whenever evidence of inefficiency or filter damage is otherwise noted.

6. Air samples shall be continuously obtained in operational areas where there is suspect or a likely potential for airborne contamination in excess of 10CFR20 limits. Exhaust systems shall be continuously sampled as proof of filter performance and to demonstrate controlled release of radioactivity.

7. A continuing program of surveillance with air and smear sampling will be conducted to detect ventilating deficiencies and to assure compliance with 10CFR20 limits.*

3.2.3 <u>Illumination</u>

Adequate facility illumination, both external and internal, and for routine and emergency situations, shall be provided for use as needed.

Proportional Counters			
Air sample counter	NMC PC-3T	$\beta-\gamma$	-
Utility counter	NMC PC-3A	$\alpha-\beta-\gamma$	-
Pulse Height Analyzer	NMC PHA-1CA	γ	-
Air Velocity Meter			
Airflow	Hastings Velometer B-16A	-	0-20,000 LFM
Portable Battery Air Samplers			
Air sampling (2)	Shop made	-	0-15 LPM
Air sampling	Gelman 15003	-	0-18 LPM
Portable Line Air Samplers			
Air sampling (5)	Gelman 13400	-	0-2.9 CFM
Air sampling (4)	Staplex TFI-A	-	0-20 CFM

Leak Detector - for hydrogen and natural gas
Davis Vapo Tester

All survey and sampling equipment is inspected and calibrated at intervals sufficient to assure reliable operation.

3.3 Operational Criteria

The operational program shall include the evaluation of releases of radioactive effluents and facility environments, the establishment of safety procedures designed to safely control contamination, personnel exposures, waste disposal, and the maintenance and operation of the necessary instrumentation and records to implement the program.

3.3.1 Basic Methods of Operations

1. *Radiological Safety Reviews and Approval

The plant procedures shall be approved radiologically safe prior to facility use. The facility operating procedures are reviewed and audited for radiological protection aspects by the Cimarron Health-Physicist, the Radiological Science Officer and the Licensing and Regulation Officer. The Cimarron Facility Manager approves the operating procedure prior to use.

The Cimarron Facility shall establish and maintain a comprehensive set of instructions for radiation health and safety practices. Such instructions shall be reviewed by the Cimarron Health-Safety Section prior to approval and issuance by the Facility Manager and annually thereafter.*

2.*Monitoring

a. Personnel monitoring shall consist of a film badge program for measuring external radiation exposures and an air sampling and bioassay program for determining internal exposure.

(1) Film badges are worn where there is potential for external radiation. The film badges will also contain an indium foil for neutron measurement and rapid determination of radiation exposure in the event of an accidental criticality. Extremity dosimeters will be worn when considered necessary to assess hand exposures. Instructions for the use of film badges or other dosimeters include the proper wearing and handling techniques to prevent spurious readings. Personnel are instructed to use the installed hand monitor upon leaving the plant operating area.

(2) Bioassay

Individuals who work routinely in areas where there is potential for internal deposition of radioactive material are checked by biomedical assay methods. Fecal analyses, whole body counting, activation blood analyses, incuded sneezing samples, swabs and other human biological assays and sampling shall be performed whenever a need is indicated as directed by the Radiological Science Officer.

A routine program for urine radionuclide analysis is designed and used to provide toxicological, radiobiological and medicolegal data on personnel. It also serves as a check on environmental measurements as well as industrial and personal hygiene. Frequency of the routine urinalysis shall be based on potential exposure levels of various work stations as indicated by air sampling results and as directed by the Radiological Science Officer.

Urinalysis will be performed at pre-employment and termination physical examinations, on persons potentially exposed to known or suspect incidents or accidental releases of radioactive material, on a spot check basis during operational periods of adequate environmental measurements or on a defined routine sampling frequency when environmental contamination is less measurable over sustained periods in relation to operational demands. Routine clinical and uranium urinalysis are performed with the consultation and approval of the Radiological Science Officer.

Follow-up analyses shall be performed as necessary to confirm normal rates of uranium elimination and absence of high chronic exposure. Results of the bioassay analyses are assembled as a composite, permanent record. The Cimarron Health-Physicist is informed of all additions to the work force, changes in work assignments, terminations of employment, and any incident which may have resulted in internal deposition to assure that appropriate schedules are maintained for bioassays.

b. Plant monitoring shall include routine and special radiation surveys of all process areas and effluents and continuous monitoring with an operational alarm system for criticality detection. The health-physics monitoring shall include radiation meter surveys, smear sampling including breathing zone sampling and effluent monitoring to insure safe radiological plant conditions. The basic objective of the air sampling monitor program shall be to obtain adequate breathing zone samples to insure and demonstrate employee protection.

Routine surveys to assure radioactive material containment are made repetitively to detect any unfavorable trends or conditions. Special surveys are conducted during maintenance of contaminated equipment. Shipments of radioactive materials are surveyed prior to release from the facility to assure that removable radioactive contamination is not significant in accordance with the applicable limits of the Department of Transportation.

Portable and permanently mounted air samplers and counting equipment suitable for detecting and evaluating the concentrations of airborne radioactive particulate will be maintained. Monitoring instruments shall be calibrated upon acquisition, after major maintenance, and at least annually.

The exhaust air and room air samplers operate on the basis of periodic removal, counting and replacement of the filter collecting media through which known air flows are drawn during sampling. Air samplers for continuous monitoring are permanently mounted at stations adjacent to suspect or troublesome containment operations as designated by the facility Health-Physicist. Samples are normally counted daily during production runs.

All new operations are initially surveyed using portable air samplers to establish a base line and to determine the need for permanently mounted stations. The frequency of sample collection and counting and locations is determined by the facility Health-Physicist. The frequency of surveys will be adequate to detect trends from the base line airborne activity established for each operation. The monitoring schedules will provide for increased sampling if the airborne activity approaches the applicable maximum permissible concentration of 10CFR20. If airborne concentrations continue in excess of the limits, the operation will be terminated until control is assured. The adequacy of air effluent control will be determined by continuous air sampling at the exhaust stacks. Liquid wastes shall be continuously sampled and analyzed to demonstrate that releases of radioactivity comply with the applicable limits of 10CFR20. Sample detection capabilities for laboratory analysis of liquid effluents shall be one tenth the allowable concentration limits of 10CFR20.*

3.*Training

In addition to a program of general industrial safety and first aid lectures, a radiological safety training program shall be in effect. All personnel shall receive an indoctrination lecture appropriate to their assigned job prior to starting work. Periodic safety training meetings shall include topics on radiation safety and will be attended by all operating personnel.*

4.*Inspections

Daily surveillance of the facility for safe operation is performed by Production Supervision.

The activites involving radioactive materials are inspected by health-safety personnel on a routine basis and the findings are reported to the Facility Manager and the Radiological Science Officer in writing each month. Conditions of an unusual or uncertain nature that could lead to problems of radiological health and safety are immediately referred to the Facility Manager for correction.

Audits of the Cimarron Facility for radiation safety and license compliance are conducted monthly by the radiological Science Officer and the Licensing and Regulation Officer. Audit findings are reported to the Facility Manager and the Division Head and items deemed to require action for correcting or improving a situation are specifically noted. A log book record of the audit findings is maintained at the Cimarron Facility. Corrective action is appropriately reported by the Facility Manager.*

5.*Personnel Protection Program

To insure that employees are working in safe conditions in the presence of radiation, the following shall be used:

a. Approved filtered and independent air supplied respiratory protective equipment is provided for non-routine operations or emergency situations and is used in accordance with license approved conditions.

b. Protective clothing and equipment is provided and worn to protect employees and minimize a spread of contamination. The amount and type of protective clothing for any specific activity is assessed on the basis of potential for personnel contamination. Available protective clothing includes lab coats, coveralls, shoe covers, gloves, goggles, safety glasses, safety shoes, respirators and shower clogs.

c. Locker room procedures are established and enforced and include proper traffic pattern to and from plant areas through showering, washing and dressing facilities.

d. Personal hygiene rules are emphasized for employee protection.

e. Personnel work rules are established and enforced to prohibit eating, smoking, or storing of food or tobacco in areas where there is a possibility for contamination with radioactive material.

f. Visitors shall be escorted in the plant processing areas.*

6.*<u>Records</u>

All plant and personnel health-safety data and reports will be recorded and filed. Timely trend analyses and reports will be made at regular intervals to management. The records of surveys and personnel exposure records are retained in accordance with applicable regulations.*

3.3.2 Emergency Planning Procedures

A system for dealing with radiological emergencies shall be set forth in the facility's emergency planning procedures. The procedures shall include evacuation plans and routes, emergency leadership organization, personnel accountability and rescue operation procedures, fire and security procedures, re-entry, decontamination, medical evacuation and hospital procedures. The emergency plan will include a list of strategic personnel and alternates that shall be notified according to procedure and will include, as the minimum, the Facility Manager, Division Head, Licensing and Regulation Officer, Radiological Science Officer, and AEC notification procedures.

3.4 Contamination-free Articles

Articles which may have been contaminated with special nuclear materials through use, handling or storage in the facility, may be disposed of or transferred to persons not licensed to possess radioactive materials when each of the following conditions are satisfied:

4.6.3 Conditions of Reflection Other than Full Reflection

Notwithstanding subsection 4.6.1 above, criticality control limits may be based on less than full reflection when each of the following conditions is satisfied:

a. The fissile material environment provides the unreflected condition or a specific degree less than fully reflected.

b. The environment of the fissile material is such that more effective reflection cannot occur through a single credible mishap.

c. The fissile material accumulation is within a structure that will preclude more effective reflection by personnel or other objects in the environment.

4.6.4 Other Moderators - If special nuclear materials are to be stored, handled or used in the presence of heavy water, beryllium or graphite, possible moderating and reflecting effects of those materials shall be considered.

4.7 Nuclear Isolation

Two accumulations or arrays of accumulations are considered as being nuclearly isolated from each other only if an edge-to-edge separation exists which is not less than one of the following or its nuclear equivalent:

1. Eight inches of water.

2. Eight inches of concrete of a density not less than 140 pounds per cubic foot. *Units stored in individual concrete isolators using an eight-inch thickness of concrete for isolation will be no more reactive than the units specified in Table IV, TID-7016(Rev.1).* For slab-type accumulations with less than six inches of air spacing between the slab face and concrete, not less than 12 inches of concrete shall be required.

3. The larger of: 12 feet or the greatest distance across an orthographic projection of either accumulation or array on a plane perpendicular to a line joining their centers.

4.8 Monitor Alarm

4.8.1 Criticality areas, monitored by systems which would not function in the event of criticality due to a loss of primary electrical power, shall be evacuated immediately upon such power loss and until power is restored and the alarm system is operative.

4.9 Methods for Establishing Limits

4.9.1 Application of Experimental Data

The available literature on nuclear criticality safety contains voluminous experimentally verified critical and safe data

Calculations involving arrays where reflection must be considered require that full array reflection be assumed. In such cases, the number of accumulations in a calculated critical array is reduced by an appropriate array reflection factor which is a constant of bare to reflected critical mass for low density reflected systems. The values of this ratio for some calculated systems to be used include:

Material	Reflection Factor
Uranium (93) metal	13
U (93) O_2	8
U (93) water mixture H/U-235 = 60	6
U (93) F_6	6
U (93) water mixture H/U-235 = 400	2.75
U (4.9) O_2F_2 Solution, H/U-235 = 500	4

The values for uranium metal and solution show sufficiently good agreement with experimental values to provide confidence in the method.

The capacity of a reflected array of unmoderated metal units calculated by the method is reduced by a factor of 2.5 for credible conditions of optimum interspersed moderation. No reduction factor need be applied over and above the normal safety margin of a maximum of 77% of the critical number of units for arrays of optimum moderated units.

4.10*Administrative Requirements

4.10.1 Nuclear Criticality Safety Reviews and Approvals

The Nuclear Safety Officer recommends, manufacturing and the Licensing and Regulation Officer review, and the Vice President, Nuclear Division, approves and modifies the criteria and standards of the Nuclear Criticality Safety Program and organization.

Calculations for criticality control limits are made by the Nuclear Safety Officer, and verified and approved by the Licensing and Regulation Officer. Operating procedures, which specify operating steps within the approved criteria and standards for criticality control, shall be written and have the prior approval of the Facility Manager. The facility operating procedures are reviewed by the Licensing and Regulation Officer for conformance to license conditions and to the program criteria and standards.*

*4.10.2 Inspections

Daily surveillance of the facility for operational safety is performed by production supervision.

The activities involving special nuclear material are inspected by Cimarron health-safety personnel on a routine basis for compliance to approved operating procedures for criticality control and the findings are reported to the Facility Manager in writing each month. Conditions of an unusual or uncertain nature relating to criticality control are immediately referred to the Facility Manager for investigation, corrective action and reporting to the Division Head.

Division level audits of the Cimarron Facility for nuclear criticality safety and license compliance are conducted monthly by the Nuclear Safety Officer and the Licensing and Regulation Officer. Audit findings are reported to the Facility Manager and the Division Head and items deemed to require action for correcting or improving a situation are specifically noted. A log book record of the audit findings is maintained at the Cimarron Facility. Corrective action is appropriately reported by the Facility Manager.

4.10.3 Records

Records of criticality control analyses shall be maintained in sufficient detail and form to permit an independent review and audit of the method of calculation and results. Such records shall be retained for a period of six months following cessation of the activities to which they apply.*

5.0 License Conditions on Packaging SNM for Transport

This section includes the technical and administrative requirements proposed for the packaging and delivery to a carrier for transport the special nuclear materials possessed under authority of License SNM-928.

The following package models are authorized for use as described herein:

Section	
1.0	Packages of UF_6 Cylinders
	1.1 Package Model No. OR-5
	1.2 Package Model No. OR-8
	1.3 Package Model No. OR-12
	1.4 Package Model No. OR-30
2.0	Package Model No. NPD-1-528
3.0	Package Model No. BE-586
4.0	Package Model No. BE-1292
5.0	Package Model No. DOT-6L (Fissile Class III Shipment)
*6.0	Package Model No. KM-1*

vessel is centered and supported in a steel angle cage, 36" x 36" x 62", covered with 16-gage sheet steel. Container constructed in accordance with Kerr-McGee Drawings NPD-205-301 (Rev.1) and NPD-1-303 (Rev.3) and subject to the following modifications:

When used for high density UO_2 pellets and the U-235 enrichment exceeds 3 w/o, the containment vessel shall be modified by inserting a 55-inch length of 8-inch diameter Schedule 20 steel pipe into the 10-inch diameter pipe.

3.2 Package Contents

3.2.1 Type and form of material: Uranium oxide, enriched to a maximum 5 w/o in the U-235 isotope, as high density pellets or dry bulk powder.

3.2.2 Maximum quantity of material per package: Three hundred (300) pounds of uranium oxide with a gross weight limit of 600 pounds for the containment vessel and its contents.

3.3 Fissile Class II and III

3.3.1 Minimum number of radiation units to be shown on label for Class II - *10.0 transport index.*

3.3.2 Maximum number of packages per shipment for Class III - 11 packages.

4.0 PACKAGE MODEL NO. BE-1292

4.1 Packaging Description

Containment vessel consists of a 5-gallon DOT Specification 37A steel pail, centered and supported within a 55-gallon DOT Specification 17H steel drum by top and bottom steel support inserts. Container constructed in accordance with Kerr-McGee Drawing No. NPD-205-302. For packaging of whole uranium buttons, a steel containment vessel constructed in accordance with the detail shown on Kerr-McGee Drawing No. NPD-205-302 shall replace the 5-gallon pail.

4.2 Package Contents

4.2.1 Type and form of material:

a. Uranium metal of any U-235 enrichment, as whole reduction buttons.

b. Uranium oxide, enriched to a maximum 5 w/o in the U-235 isotope, as dry bulk powder, or pelletized material.

4.2.2 Maximum quantity of material per package:

a. For the contents described in 4.2.1a above:

Fourteen (14) kilograms U-235

b. For the contents described in 4.2 lbs above:

Ninety (90) pounds maximum weight of contents and for the following w/o U-235, the U-235 content for the form specified shall not exceed the following:

*Maximum w/o U-235	Maximum U-235 (kgs.) as pellets	Maximum U-235 (kgs.) as powders
5.0	0.657	0.774
4.5	0.723	0.896
4.0	0.792	1.021
3.5	0.837	1.116
3.0*	0.904	1.233

4.3 Fissile Class II and III

4.3.1 Minimum number of radiation units to be shown on label for Class II:

a. For the contents described in 4.2.1a and limited in 4.2.2a:

3.2 transport index

b. For the contents described in 4.2.1b and limited in 4.2.2b:

3.9 transport index

4.3.2 Maximum number of packages per shipment for Class III:

a. For the contents described in 4.2.1a and limited in 4.2.2a:

33 packages

b. For the contents described in 4.2.1b and limited in 4.2.2b:

26 packages

6.0 PACKAGE MODEL No. KM-1

6.1 Packaging Description

The package is a trailer assembly consisting of a fabricated slab structure which is bolted to the frame of a specially designed 40-foot trailer. The package is constructed and assembled to form a slab structure (4.5 ± 0.1 inches high x 7 3/4 feet wide x 40 feet long) for containment of the product in transport. Details of construction are shown on Kerr-McGee Drawings NDA-2-501-17, Sheets 1 through 5, Pellet Shipping Trailer.

6.2 Package Contents

A maximum of 15,000 pounds of UO_2 pellets of enrichments not exceeding 5% U-235. The slab containment vessel will be a subcritical infinite slab thickness for the enrichment being transported assuming optimum moderation of the contents with the package closely reflected on all sides by water.

The inside dimension of the slab containment vessel shall not exceed the maximum slab thickness for the enrichment being transported in accordance with the following:

Enrichment (% U-235)	Maximum Slab Thickness (inches)
≤3.5	4.6
3.51 to 4.30	4.1
4.31 to 5.0	3.7

6.3 Fissile Class III shipments only by motor vehicle.

6.4 Supervisory inspection of the slab thickness shall be made prior to each loading of the package to assure compliance for the enrichment to be transported.

BASES FOR

LICENSE CONDITIONS

FOR THE

NUCLEAR DIVISION CIMMARON FACILITY

APPENDIX D

License SNM-928

Docket No. 70-925

APPENDIX D

BASES FOR LICENSE CONDITIONS

This Appendix D is a compendium of the rationale for the establishment of the license conditions which are given in Appendix A. These bases for the license conditions provide for concise interpretation of the administrative and technical specifications as an aid in compliance evaluations. The bases are further intended as a reference to assist in fulfillment of the license conditions and provide guidance in considerations of proposed changes to the license conditions.

The bases for license conditions shall be maintained to reflect current license authorizations.

1.1 Possession Limits

(a) 3,000 kilograms of U-235 is deemed to be the maximum quantity in inventory required for possession at any one time within the confines of the Cimarron Facility. Consideration is given to realistic production capacities. A breakdown of the anticipated material inventory for any particular time at maximum production activity would be approximately 1,000 kg U-235 contained in high enriched uranium (>5% U-235) and about 2,000 kg U-235 contained in low enriched uranium (<5% U-235).

(b) A possession limit of 500 grams U-235 at the Kerr-McGee Research Center will accommodate the requirements for the special analytical services performed there on a routine basis. This single subcritical mass limit precludes further consideration for nuclear criticality safety. Compliance to the possession limit is administratively controlled by maintenance of a running inventory.

1.2 Authorized Activities

The Cimarron Facility of the Kerr-McGee Nuclear Division is a nuclear fuel conversion facility with limited fuel fabrication capabilities designed to meet the needs of an expanding nuclear fuel market. Excluded from the current Cimarron Facility scope are fundamentally different activities such as the possession and processing of plutonium and U-233, irradiated fuel materials handling and fabrication of reactor fuel bundles and fuel elements.

The authorized activities represent current facility capability and demonstrated competence of operational safety in conducting such activities. The enriched uranium materials at the Cimarron Facility are processed and controlled on a job shop basis. Flexibility for process changes, equipment movements and operational area modifications as required for production needs are implied.

Activities which are not included in the authorized list would not be conducted without prior AEC approval by means of license amendment. Improvements in the authorized activities are permitted after internal review and approval in accordance with the applicable license conditions.

2.0 Organization, Administration and Personnel

Section 2 of Appendix A contains the fundamental requirements of organizational structure, staffing, position responsibilities and personnel qualifications. These requirements provide for effective administration of the functional programs relating to the regulatory elements of radiation protection and nuclear criticality safety.

The requirements are deemed by Nuclear Division management as necessary to provide reasonable assurance for the safe conduct of the activities authorized for the Cimarron Facility. Decisions on matters of safety functional control are made by the Division Head incorporating the advice and recommendations of professional specialists. Organization components in the Nuclear Division are established, defined and maintained to provide competent facility management, audits and surveillance to assure compliance and to measure effectiveness of the health-safety programs, and to reasonably minimize the reporting lines for maximum effectiveness in functional control.

The Nuclear Division shall maintain within its organization sufficient technically qualified individuals to analyze proposed changes and to establish safe practices and control limits for the operational changes.

Radiological Protection

The Nuclear Division Manufacturing Department has the responsibility for operation of the Cimarron Facility. The intent in describing the radiological protection functional organization is to emphasize the established independence for overall safety from operating pressures and to itemize the principal responsibilities for positions involved in the program to assure protection against radiation.

Nuclear Criticality Safety

To accommodate the nuclear criticality safety needs of the expanding activities at the Cimarron Facility, two staff positions within the Nuclear Division have designated responsibilities in the nuclear criticality safety functional organization. The positions are filled by criticality safety specialists to provide the criticality controls for the Cimarron Facility independent of operational responsibility and to audit compliance. Currently, in view of the limited experience of incumbents of the Cimarron Health-Safety Section, all analyses for criticality control are performed, verified and approved by the Division Nuclear Safety Officer and the Licensing and Regulation Officer. Such approvals are made in writing, preceded by telephone approval, if required for timeliness. Visual inspections are made as appropriate to properly assess the environmental factors that must be considered in the nuclear safety analyses for proposed changes.

3.0 Radiological Protection Program Specifications

The license conditions regarding radiological protection represent the requirements which are considered vital for safe conduct of the activities. As a minimum, these program elements will be maintained in effect for protection against radiation during the conduct of the activities authorized for the Cimarron Facility.

The general program elements include the basic essentials for a sound radiological protection program. These requirements represent a consideration of regulatory requirements and Kerr-McGee's acceptance of its responsibility to provide for the safety of the employees and individuals in the surrounding community of the Cimarron Facility.

3.1 Facility Design

Safety is given first consideration in all company activities. The bases for design of the Cimarron Facility included an overall consideration to provide for effective effluent and radioactive contamination control. Site selection similarly was made to minimize the exposure potential of the public to radioactivity in the event of an accidental airborne release or criticality incident.

3.2 Equipment

3.2.1 The primary design basis for process equipment including hoods, gloveboxes and enclosures is to provide containment of radioactive materials for the prevention of air contamination in areas occupied by personnel.

3.2.2 Ventilation and Exhaust System

1. The requirement that the ventilation equipment provide an air flow from areas of lesser contamination to areas of higher contamination is to prevent contamination spread.

2. The potential for accumulation of radioactive dust in exhaust ducts is to be eliminated as much as practicable. The objective is to minimize cleanup after a production run and to prevent undue exposure hazard during maintenance on exhaust systems.

3. The ventilation and exhaust system designs are to provide operational areas which do not exceed an airborne concentration of radioactivity of 6 x 10^{-11}uc/ml during routine operations. Allowance for personal respiratory protection is not to be taken for routine operations.

4. The basis for the requirement that open face hoods maintain an air flow of at least 100 lineal feet per minute is to prevent a spread of radioactive material from the containment area to the operating area.

5. The exhaust systems for airborne effluents are equipped with high efficiency absolute filters to minimize operational losses of material and to comply with regulatory limits by controlling the release of radioactivity to the unrestricted environment. Differential pressures across filters are checked to assure that filters are not plugged or damaged and are functioning in the proper manner.

6. The basis for requiring continuous air sampling on exhaust stacks is to assure that the filters remain efficient.

3.2.3 Illumination

Adequate facility lighting in the plant is provided for routine operation to maintain a favorable plant environment which is conducive to safe operating practices. An emergency internal lighting system which is battery operated is provided for obvious safety reasons to perform necessary operations in the event of an electrical power failure. An external plant lighting system is provided for normal, industrial security purposes.

3.2.4 Emergency Equipment

The emergency equipment available for use shall be adequate to effectively cope with the emergency situations which are considered credible and/or likely at the Cimarron Facility. Equipment to reasonably handle fires, accidents, a criticality incident and medical first aid cases are considered in determining the extent and type of emergency equipment.

3.2.5 Fire Fighting Equipment

The number and placement of carbon dioxide fire extinguishers in the plant is based on local building codes. Other fire fighting equipment shall be adequate to provide reasonable assurance that a facility fire emergency can be locally controlled until assistance arrives from the nearby fire department if so needed.

3.2.6 Safety Education Equipment

The basis for providing safety education aids lies in the consensus of safety personnel that such measures help promote safety mindedness and foster accident prevention.

3.2.7 Health-Physics Monitoring Equipment

Radiation detectors, air samplers and radiation counting equipment shall be in adequate supply to conduct the surveys necessary to demonstrate compliance with the regulatory requirements of 10CFR20.

A gamma detector capable of measuring 1000 R/hr is made available to effect re-entry in the event of a criticality accident. Beta and gamma detectors are available for routine plant surveys. Alpha detectors are provided to aid in contamination control.

3.3 Operational Criteria

3.3.1 Basic Methods of Operation

1. Radiological Safety Reviews and Approval

The broad licensing concept acknowledges that the competent licensee organization is capable of improving and adjusting operations without compromising safety by equipment modifications and operating procedure changes.

The philosophy on operational safety within the Kerr-McGee Nuclear Division assigns principal responsibility to the Facility Manager with support by staff safety functions. The bases for the review and approval system is the recognition that the Cimarron Facility requires some operational flexibility with the Facility Manager technically competent to properly decide on changes involving safety criteria and standards and to refer proposed changes to the appropriate specialists for review and approval.

2. Monitoring

The film badge program is adopted to measure external radiation exposures. Indium foil is used for criticality dosimetry. The bioassay requirements are based on a need for surveillance of personnel regarding internal deposition of radioactive material beyond the air sampling program. Radiation surveys and smear sampling are intended to aid in contamination control and are considered a supplement to the air sampling program.

3. Training

The basis for radiological safety training is the prevention of undue radiation exposure through an understanding of fundamentals of radiation hazards and controls.

4. Inspections

The basis for the inspection requirements is reasonable assurance that an unsafe plant practice does not persist.

5. Personnel Protection Program

The bases for the elements of the personnel protection program are the prevention of undue personnel contamination and spread of radioactive contamination outside the confines of the facility.

6. Records

The maintenance and retention of radiation survey and personnel exposure records is based on requirements of applicable regulations. The basis for health-safety reports is to keep management routinely informed of favorable or unfavorable trends in plant safety conditions, health-safety problems and apprised of the effectiveness of the radiation safety program.

3.3.2 Emergency Planning Procedures

The basis for facility emergency planning and written procedures is to minimize personnel injury and to avert unnecessary property damage.

3.4 Contamination-Free Articles

The basis for the decontamination requirements on articles which may be released to non-licensees is that the remaining level of contamination is insignificant and cannot be injurious to the general public regardless of subsequent usage.

3.5 Respiratory Protection Allowance

The bases for the respiratory protection program are given as objectives in Section 3.6.1 of Appendix A.

4.0 Nuclear Criticality Control Specifications

The basis for the program elements is the prevention of nuclear criticality at the Cimarron Facility.

4.1 Basic Requirements

4.1.1 A recognition of the potential hazard and serious consequences of an accidental criticality at the Cimarron Facility demand that a process analysis be performed before beginning an operation with fissionable materials or changing an existing operation. It shall be determined that the entire process will be subcritical under both normal and credible abnormal conditions.

4.1.2 The bases for establishing subcriticality are to be noted for all significant conditions in the process.

4.1.3 Reliance upon operating personnel judgments or actions for criticality control is not acceptable assurance against the consequences of an accidental criticality.

4.1.4 The application of limits based on experimental data provide the greatest degree of confidence presently attainable in nuclear safety technology and experimental data shall be used for criticality control wherever such limits are not detrimental to operational needs.

4.1.5 The established check and balance system governing operating procedures provides reasonable assurance that adherence to the procedures shall prevent a criticality incident.

4.1.6 The basis for nuclear criticality safety training of operating personnel is to foster an awareness of the consequences of accidental criticality and to develop personal attitudes of responsibility toward job performance.

4.1.7 Operations shall be reviewed frequently to ascertain that procedures are being properly followed and that process conditions have not been altered so as to affect the nuclear criticality safety evaluation. The basis for surveillance and audit requirements is to provide reasonable assurance that unsafe practices or suspect situations will be timely detected and corrective action initiated as required.

4.2. Specific Operational Requirements

4.2.1 The basis for the double contingency rule is to assure that an additional safety factor is constantly present to avoid nuclear criticality even though an unlikely change occurs in the system.

4.2.2 Geometry limitation is chosen as the primary selection for nuclear safety control as it is generally considered to offer a more difficult-to-change control over other available parameters. Full advantage may be taken of any nuclear characteristics of the process materials and equipment. Control shall be exercised to maintain all dimensions and nuclear properties on which reliance is placed.

4.2.3 Single-parameter limits are generally adequate for many of the Cimarron Facility operations and are more easily applied and controlled than multiparameter limitations.

4.2.4 The generalization that assumptions of optimum moderation and full water reflection shall be used whenever possible to set criticality limits is provided to take advantage of any free contribution to criticality control.

4.2.5 The activities at the Cimarron Facility involve both heterogeneous and homogeneous fissile material systems and control must be exercised over both the mass and distribution to assure nuclear criticality safety.

4.2.6 Stringent control is required to assure a continued presence with the intended distributions and concentrations of neutron absorbers. When used for primary control against criticality, exercising such control is difficult. The advantage of use of neutron absorbers for primary control has not been obvious in the Cimarron activities.

4.3. Maximum Permissible Values for Isolated Units

The maximum permissible values for single accumulations are obtained by reducing the critical values by the application of the appropriate safety factor. The basis for the safety factors of Section 4.3, Appendix A, is to incorporate in the process specifications margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.

The requirement that accumulations limited by reactivity shall not exceed 90% of critical is to provide adequate safety margin for calculated uncertainty for conditions of optimum moderation and full water reflection.

4.4 Maximum Permissible Values for Interacting Units

The maximum permissible values for interacting units of fissile accumulations when the critical number is calculated are obtained by reducing the critical number by the application of the appropriate safety factors. The basis for the safety factors of Section 4.4, Appendix A, is to incorporate adequate margins against calculational uncertainties.

4.5 Integrity of Structures

Structures used to maintain criticality control are designed on the basis of good engineering practices.

4.6 Conditions of Moderation and Reflection

4.6.1 Optimum Moderation and Reflection

Advantage is to be taken of any free contribution to criticality control whenever possible.

4.6.2 Conditions Other Than Optimum Moderation

Certain plant operations are nuclearly safe with moderation conditions being other than optimum. To assure and demonstrate criticality control, such fissile material systems are evaluated for moderation conditions more nearly representative of the actual system. The basis for conditions of less than optimum moderation is primarily the absence of moisture in the fissile material accumulation required by the normal process specifications. Full water reflection is assumed in establishing limits which are moderation controlled. The basis for undermoderated limits for criticality control is that no single credible mishap will result in a more effective degree of moderation than assumed in the specific criticality analysis.

The basis for overmoderated systems is reasonable assurance that an increase in the concentration of the fissile material shall not occur with normal operating conditions or likely process changes.

4.6.3 Conditions Other Than Full Reflection

Certain plant operations are nuclearly safe with conditions of less than full water reflection. To assure and demonstrate criticality control such fissile material systems are evaluated for the credible reflection situation. The basis for conditions of less than full water reflection for criticality control is that no single credible mishap will cause a more effective reflection than assumed in the specific criticality analysis.

4.6.4 Other Moderators

The moderating and reflecting effects of heavy water, beryllium and graphite, influences the critical values of fissile material systems to warrant out-of-ordinary or special consideration. The basis for this license condition is to assure that proper consideration be given in the unlikely event that such reactor materials be encountered.

4.7 Nuclear Isolation

Experimental data and extrapolation of data obtained from experiments is the basis for the isolation distances and thicknesses of isolation materials specified in the license condition covering nuclear isolation. (Refer TID-7016 (Rev.1), ORNL-2367 and BNWL-SA-268).

4.8 Monitor Alarm

The criticality detectors and alarm system are provided in accordance with regulatory requirements to warn employees of the occurrence of a nuclear criticality. The risk of employee exposure to a potential criticality is permitted only if warning protection is provided by an operative detection-alarm system.

4.9 Methods for Establishing Limits

4.9.1 Application of Experimental Data

The proper application of experimental data for criticality control is basic to a sound nuclear criticality safety program. Process specifications shall incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.

4.9.2 Calculative Methods

Criticality analyses should be based on calculative methods which provide satisfactory agreement between calculated values and experimental data. The calculative methods authorized for use by the Kerr-McGee Nuclear Division have been shown during the methods development and on a continuing basis, by nationally recognized criticality specialists, to yield conservative results. The methods have found widespread acceptance as satisfactory calculative techniques for criticality control.

4.10 Administrative Requirements

4.10.1 Reviews and Approval

The established check and balance system for nuclear criticality safety is deemed by Nuclear Division management as necessary to provide reasonable assurance that a nuclear criticality will not occur at the Cimarron Facility.

The Vice President, Nuclear Division, is responsible for all activities of the Kerr-McGee Nuclear Division. The responsibility is delegated to the Cimarron Facility Manager to recognize new operations and changes in operations which can affect nuclear criticality safety. He is accountable to obtain the necessary criticality analyses and operating procedures prior to adopting new or modified process changes which could require a change in the criticality controls.

4.10.2 Inspections

The basis for these license conditions is reasonable assurance that unsafe or suspect practices and deviations from control procedures will be timely detected and investigative and corrective action taken to avoid a criticality incident.

4.10.3 Records

The maintenance of records of criticality control analyses is required to assure that the pertinent information is available to allow a full investigation in the event of a criticality incident. Such records also serve to demonstrate compliance to other relevant license conditions.

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