769

KY-550

AEC RESEARCH AND DEVELOPMENT REPORT

MASTER

DEMONSTRATION FUEL ELEMENT SHIPPING CASK

FROM

LAMINATED URANIUM METAL

PART I

GENERAL INFORMATION, BACKGROUND, AND DESIGN

C. W. LOVELAND

D. H. STITT

S. BERNSTEIN

UNION CARBIDE CORPORATION
NUCLEAR DIVISION
PADUCAH GASEOUS DIFFUSION PLANT

UNION
CARBIDE

PADUCAH GASEOUS DIFFUSION PLANT
P. O. Box 1410
Paducah, Kentucky 42001

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

Printed in the United States of America

Available from

Clearinghouse for Federal Scientific and Technical Information

National Bureau of Standards, U. S. Department of Commerce

Springfield, Virginia 22151

Price: Printed Copy \$3.00; Microfiche \$0.65

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

Date of Issue: April 15, 1969

Report Number: KY-550

Subject Category: UC-2

GENERAL, MISCELLANEOUS,
AND PROGRESS REPORTS
(TID-4500, 53rd Ed.)

DEMONSTRATION FUEL ELEMENT SHIPPING CASK

FROM LAMINATED URANIUM METAL

PART I

GENERAL INFORMATION, BACKGROUND, AND DESIGN

C. W. Loveland

D. H. Stitt

S. Bernstein

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

UNION CARBIDE CORPORATION
NUCLEAR DIVISION
Paducah Gaseous Diffusion Plant
Paducah, Kentucky

Report Number: KY-550

Subject Category: UC-2

GENERAL, MISCELLANEOUS,
AND PROGRESS REPORTS
(TID-4500, 53rd Ed.)

Authors:

C. W. Loveland
D. H. Stitt
S. Bernstein

ABSTRACT

A demonstration fuel element shipping cask that uses laminated uranium metal as the shielding material was designed, fabricated, and tested by the Paducah Gaseous Diffusion Plant, operated by Union Carbide Corporation, Nuclear Division, for the U. S. Atomic Energy Commission. This is the first of a series of three reports covering the total project. This report discusses the background of the program, presents general information concerning pertinent chemical and physical properties of uranium related to its use for shielding applications, and covers the design and engineering phases of the project. A complete set of the design drawings of the cask is also included. The overall project demonstrated the feasibility of using laminated uranium metal for radioactive material shipping casks.

TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT	i
LIST OF FIGURES	v
LIST OF TABLES	vii
INTRODUCTION	1
SUMMARY	2
BACKGROUND	3
Historical	3
Paducah Plant Experience	4
Uranium Technology Considerations	5
Diffusion Barrier Development	7
Grain Randomization Development	8
Uranium Welding Development	13
Chemical Activity Control Development	14
PROGRAM OBJECTIVES	16
PROGRAM CRITERIA	19
Selection of Basic Shape and Size	19
Selection of Indian Point Core A Cask for Comparison	20
Basic Criteria	20
DESIGN	24
Conceptual Design	24
Response of a Laminated Uranium Cylinder to Mechanical Shock	24
Thermal Conductivity Determination	26
Thermal Design of the Conceptual Cask	26
Mechanical Design of the Conceptual Cask	36

Phase I Design	38
Development of Final Design	38
Quarter Scale Model	40
Phase II Design	40
Free Fall Analysis	40
Closure Redesign	44
Design Summary	48
APPENDIX A: Uranium Metal Information Pertinent to its Use for Radiation Shielding	51
Physical Properties	51
Thermal Cycling	51
Oxidation	51
Uranium Availability	56
Effect of a Depleted Uranium Shield on the Criticality of Irradiated Fuel	56
Uranium Metal Health Hazard Control	56
APPENDIX B: Factors Affecting the Weight Ratio of Uranium to Lead for Radiation Shielding	59
APPENDIX C: Alloying of Uranium and Stainless Steel in Uranium Shielded Casks	61
APPENDIX D: Engineering Drawings of Demonstration Uranium Fuel Element Shipping Cask	67
REFERENCES	85
ACKNOWLEDGEMENTS	87

LIST OF FIGURES

		<u>Page</u>
Figure 1	Prototype Laminated Uranium Metal ^{60}Co Shipping Cask.	6
Figure 2	Uranium Growth Test.	12
Figure 3	Open-Ended Laminated Cylinder Used for Impact Testing	25
Figure 4	Open-Ended Laminated Cylinder After Corner Drop.	27
Figure 5	Open-Ended Laminated Cylinder After Side Drop.	28
Figure 6	Sections Cut from Open-Ended Laminated Cylinder Following Testing.	29
Figure 7	Section Through Laminated Uranium 30-Inch O.D. Cylinder Used for Fin Design	30
Figure 8	Convective and Radiative Heat Transfer Coefficients as a Function of Temperature Difference When Ambient Temperature is 130°F	32
Figure 9	Fin Spacing on Demonstration Cask.	33
Figure 10	Surface Temperature as a Function of Surface Heat Flux When the Ambient Temperature is 130°F	35
Figure 11	Conceptual Design of the Demonstration Fuel Element Shipping Cask.	37
Figure 12	Phase I Design of the Demonstration Fuel Element Shipping Cask.	39
Figure 13	Original Quarter Scale Model of the Demonstration Fuel Element Shipping Cask.	41
Figure 14	Impact Fins.	43
Figure 15	Calculated Average and Peak Acceleration Curves for Demonstration Cask 30-Foot Free Fall Test.	45
Figure 16	Phase II Design of the Demonstration Fuel Element Shipping Cask.	47
Figure A-1	Thermal Cycling Range and Temperatures Effect on Growth of Uranium	55

Figure C-1	Alloying Text Fixture	63
Figure C-2	Diffusion Barrier Test Sample	64
Figure C-3	Diffusion Barrier Test Sample	64
Figure C-4	Diffusion Barrier Test Sample	65
Figure C-5	Diffusion Barrier Test Sample	65
Figure D-1	Demonstration Uranium Fuel Element Shipping Cask . .	67
Figure D-2	Demonstration Uranium Fuel Element Shipping Cask- Cask Assembly	69
Figure D-3	Demonstration Uranium Fuel Element Shipping Cask- Cask Sections and Details	71
Figure D-4	Demonstration Uranium Fuel Element Shipping Cask Cover Assembly	73
Figure D-5	Demonstration Uranium Fuel Element Shipping Cask Lifting Device	75
Figure D-6	Demonstration Uranium Fuel Element Shipping Cask Cover Details	77
Figure D-7	Demonstration Uranium Fuel Element Shipping Cask Skid Assembly	79
Figure D-8	Demonstration Uranium Fuel Element Shipping Cask Skid Sections and Details	81
Figure D-9	Demonstration Uranium Fuel Element Shipping Cask Strap Details	83

LIST OF TABLES

	<u>Page</u>
Table 1 Thermal Expansion Coefficients for Various Materials	9
Table 2 Comparison of Two Existing Fuel Element Shipping Casks.	21
Table A-1 Physical and Chemical Properties of Some Shielding Materials	52
Table A-2 Broad Beam Gamma-Ray Attenuation.	53
Table A-3 Comparison of Lead and Uranium Cylindrical Cask Shields.	54
Table A-4 Effect of Temperature on Corrosion of Uranium in Air.	54
Table B-1 Effect of Shape Factor on Uranium-Lead Weight Ratio for Shielding.	60

INTRODUCTION

A demonstration fuel element shipping cask utilizing laminated uranium metal as the shielding material was designed, fabricated, tested, and evaluated by the Paducah Gaseous Diffusion Plant, operated by Union Carbide Corporation, Nuclear Division, for the U. S. Atomic Energy Commission. The project was authorized by the AEC Division of Reactor Development and Technology.¹ This report, covering the design and engineering of the demonstration cask, is the first in a series of three reports covering the overall project. The background of the program, including a description of some of the pertinent chemical and physical properties of uranium for this application, is included. Other reports in this series will cover (a) the fabrication of the demonstration cask and the economic factors relative to the fabrication and use of laminated uranium metal shipping casks² and (b) the testing and demonstration phases of the project.³

Specific basic objectives were determined as a guide to the overall program. It was decided to build a demonstration cask of practical design, weighing approximately 12 tons, utilizing the same performance criteria as an existing lead cask. Accordingly, the demonstration cask was designed to be a prototype of a cask which could carry four Indian Point core A spent fuel assemblies except that its length would be shortened to reduce weight and to minimize costs. If the demonstration cask had been built full scale, it would have weighed approximately the same as the Indian Point lead cask; however, it would have had twice the payload capacity.

The design goal of incorporating all program criteria into a functional cask required the generation of previously nonexistent information concerning unique aspects of fabrication and operation of a laminated uranium shielded cask. Investigation of factors related to the operation of a completed cask, such as heat transfer and impact resistance, revealed the need for development of information concerning the response of a laminated uranium structure to mechanical and thermal loading. This information was developed in a series of related tests. The final cask design was based on a rigid shield, sacrificial external-member concept, in which the uranium shield would support fins and other extended stainless steel members that could be destroyed in an accident without compromising the integrity of the cask. As a result of analysis and model testing, it appeared certain, prior to testing, that the cask would meet all the regulations. This determination was subsequently substantiated by the testing program.

SUMMARY

The desirability of developing spent reactor fuel shipping casks which would provide the ultimate in public safety without being prohibitively expensive to fabricate or transport resulted in the authorization of a project to demonstrate the feasibility of designing, fabricating, and testing a demonstration radioactive material shipping cask from laminated uranium metal at the Paducah Gaseous Diffusion Plant where the technology for uranium metal fabrication exists.

Previous attempts to use uranium for shielding purposes had been generally unsatisfactory due to the lack of an adequate fabrication technology. Prior to this project, significant technological improvements had been made in the following areas: development of a diffusion barrier between uranium and stainless steel to prevent formation of a low melting point alloy at fire temperatures; development of a practical procedure to prevent any possible occurrence of thermal cycling growth in the rolled uranium plates; optimization of uranium welding techniques to minimize fabrication costs while simultaneously providing welds of the highest quality; and development of procedures and methods to satisfactorily control the chemical reactivity of uranium.

These achievements provided the technology necessary for the design and fabrication of a demonstration cask which subsequently met all of the packaging standards and hypothetical accident criteria specified in the applicable regulations.

BACKGROUND

HISTORICAL

The Advisory Committee on Reactor Safeguards has been actively following the problems associated with the development of large spent fuel shipping casks. The Advisory Committee undertook a thorough review of the problem several years ago and recommended that the research program on spent fuel shipping casks include the study of shielding materials other than lead. There is concern about the possible shortcomings of lead under accident conditions involving fire or impact. The high coefficient of expansion of lead combined with its relatively low melting point can lead to a loss of shielding during fire, and lead can cold flow at the time of impact from a fall which could result in a void space within the cask and a resultant loss in shielding. Since that time, reservations have also been expressed concerning the use of lead casks to transport spent fuels from future generation power reactors where the greatly increased decay heat load generated by the spent fuel may result in internal cask temperatures which exceed the melting point of lead. The recent dramatic surge in the growth of the nuclear power industry has served to place further importance on the development of spent reactor fuel shipping casks which will provide the ultimate in public safety without being prohibitively expensive to fabricate or transport.

To date, essentially all large casks used or contemplated for use in the United States are made of lead encased in steel. Other materials such as iron, steel, wood, and tungsten have been considered for cask fabrication; however, excessive costs or resultant large increases in size and weight of casks have minimized any extensive use of these materials.

The use of uranium metal as a shielding material for shipping casks appears desirable because of its high density, radiation absorption efficiency, thermal conductivity, high melting point, high strength, and availability. A discussion of some of the information on uranium metal which is pertinent to its use for radiation shielding is presented in appendix A. A brief summary of the more important factors is presented below.

Because of its high density (~ 19 g/cc), a cylindrical uranium shield will usually weigh less than 75 percent of a lead shield and less than 50 percent of a steel shield having comparable radiation attenuation. The actual ratio of the weight of uranium to the weight of lead in any shield depends upon the specific shape of the shielding device. This relationship is discussed in greater detail in appendix B.

Uranium has a high thermal conductivity and specific heat. The thermal conductivity is about the same as solid lead. Uranium has a melting point of 2071°F as compared to 621°F for lead.

Although uranium oxidizes upon exposure to air in a manner similar to the rusting of mild steel, canning in stainless steel or the application of a protective coating which prevents contact between uranium and air effectively eliminates any oxidation problem.

A question is frequently raised concerning whether the fissile content, i.e., ^{235}U , of a depleted uranium shield would increase the nuclear safety problems associated with the shipment of irradiated fuel elements. Analysis indicates that the low ^{235}U content of the shield material ($\sim 0.2\%$) would not increase these problems. In addition, experimental results of tests indicate the neutron reflective properties of lead, uranium, and water to be similar.

Uranium can be rolled, formed, machined, and welded with the same type shop equipment that is used to fabricate mild steel. Although many of the structural properties of uranium are approximately the same as mild steel, the differences in chemical properties between the two materials require the use of special procedures and techniques. These procedures and techniques have been developed by AEC contractors working with uranium; however, they could also be readily used by industrial fabricators.

Since uranium may have a significant weight and volume advantage when compared to lead, there should be a significant savings in shipping costs when transporting uranium casks. These savings would occur not only from reduced transportation charges but also from more economical construction of specialized transportation carriers. Shappert and Salmon⁴ compared shipping costs for lead, steel, and uranium shielded casks on the basis of the same cask weight and based on the total cost for transporting spent fuel from a reactor to a reprocessing site 1000 miles distant. The results of their study showed that the use of uranium shielded casks can be economically advantageous provided that the cost of the cask can be kept within a range of \$4 to \$9 per pound of cask weight (empty) based on a lead cask cost of \$1 per pound. Current indications are that by 1980 the domestic nuclear power generation will be at least 150,000 mwe. Thompson and Morgan's⁵ review of nuclear transportation indicated that for the extremely conservative estimate of a nuclear power economy of 70,000 mwe an annual savings of \$50 million in shipping charges may be attainable by economies resulting from transportation research. The demonstration uranium shipping cask program is one phase of the overall effort being conducted by the AEC to develop the technology required to support the projected nuclear power industry.

PADUCAH PLANT EXPERIENCE

Experience obtained in the operation of a uranium metal production facility and in the fabrication of uranium components for special applications has resulted in a staff at the Paducah Plant who are familiar with the unique physical and chemical properties of uranium metal. One of the fabrication procedures developed consists of producing thick uranium cylinders by shrinking together successive cylinders made from rolled uranium plate. A thick cylinder produced in this manner has physical properties similar to a cast, monolithic structure; and the absence of any void areas, such as can develop during solidification of large castings, can be assured. It was postulated that a shipping cask which used laminated uranium as its radiation shield would be superior to casks built with lead for reasons noted previously, i.e., improved resistance to deformation upon impact, improved resistance to fire, reduced weight, etc.

Accordingly, during FY-1966, a proposal was made to the AEC for the Paducah Gaseous Diffusion Plant to build a demonstration radioactive material shipping cask from laminated uranium metal to show its feasibility. While the proposal was being considered by the AEC, discussions were held with personnel from the Isotopes and Chemical Technology Divisions at the Oak Ridge National Laboratory (ORNL) regarding the use of this technology to fabricate high capacity radioisotope shipping casks. It was subsequently agreed that Paducah would design and fabricate a prototype laminated uranium metal shipping cask for ORNL, weighing approximately 10,000 pounds, for transporting large quantities of ^{60}Co .

Figure 1 is a photograph of this cask. Details applicable to the design, fabrication, and performance of this unit have been reported elsewhere.⁶ Technological progress was achieved in several areas while fabricating the prototype ^{60}Co cask, and it was subsequently decided to build for the ORNL Isotopes Division an advanced prototype cask capable of transporting 400,000 curies of ^{60}Co in a single shipment. Fabrication of this unit was completed in FY-1967. Performance of these units has been satisfactory, and Department of Transportation approval (S.P. No. 5595) has been obtained for their use in international shipments.

In June 1966, the AEC Division of Reactor Development and Technology authorized the Paducah Plant to carry out the proposed demonstration fuel element shipping cask project. By the time the detailed design work was required for the demonstration cask, the Paducah Plant had designed and fabricated two large isotope shipping casks, and the technology developed while doing this work was incorporated into the demonstration cask design.

URANIUM TECHNOLOGY CONSIDERATIONS

Prior to the inception of the current program at Paducah, uranium had not been seriously considered for use in shielding applications, primarily because of a general lack of knowledge relative to its availability and properties. A few attempts to use uranium for shielding purposes had been made; however, the results were generally unsatisfactory due to the absence of an adequate fabrication technology.

Uranium can be fabricated by using relatively unsophisticated methods; however, this metal has unique properties which require specific procedures in order to obtain the desired properties in the finished item. Many of the properties required for radiation shielding assemblies are different from those related to weapons or reactors, and the necessary technology to obtain these properties has heretofore been nonexistent. Some of these properties which are factors in the design and fabrication of laminated uranium shipping casks, and the procedures which were developed to be compatible with these properties, are discussed in the following paragraphs. Included are:

1. Uranium in contact with stainless steel can form a low melting point alloy (m.p. 1337°F) if heated to fire temperatures.

PROTOTYPE LAMINATED URANIUM METAL
 ^{60}Co SHIPPING CASK

Figure 1

3095-12 (U)

2. Uranium plate in the as-rolled condition will undergo a permanent change in physical measurements when subjected to certain thermal cycling conditions. As-rolled uranium plate also differs from most materials in that it exhibits different coefficients of thermal expansion in different planes, with the coefficient in one axis actually being negative.
3. Uranium is difficult to weld due to its chemical reactivity. Hot uranium metal oxidizes very rapidly in air unless special procedures are used to protect the metal surface.
4. Uranium metal reacts chemically with water, oxygen, or hydrogen to form compounds which could be detrimental to the integrity of uranium shipping casks.

Diffusion Barrier Development

As currently visualized, shipping casks using uranium will be clad both internally and externally with stainless steel. It was known that stainless steel can form a low melting point alloy with uranium by diffusion at fire temperatures, and it was feared that this reaction could contribute to a shielding failure if the cask were subjected to a fire. Because of this possibility, a program was conducted to find a suitable method to prevent the formation of a low melting point alloy between stainless steel and uranium at fire temperatures.

The formation of low melting temperature alloys by atomic diffusion at temperatures below the melting point of either component is not a rare phenomenon. The basic factors involved in this type of reaction are well understood and are discussed very thoroughly in the literature.⁷ A report covering the initial laboratory-scale investigation conducted by the Paducah Plant Laboratory Division to find a method for preventing possible low melting point alloy formation between uranium metal and cask cladding during exposure to fire is presented as appendix C.

As a result of the initial Paducah laboratory work, a 5-mil copper diffusion barrier was applied between the cladding and the outer uranium lamination on the prototype large isotope shipping cask.⁶ The barrier was flame sprayed on the stainless steel surface prior to the cladding being placed against the uranium. In the advanced prototype large isotope shipping cask a similar diffusion barrier was used; however, in this case 5 mils of tungsten were plasma sprayed on the inside surface of all stainless steel cladding. In a laboratory-scale test, a block of uranium 6-in. by 12-in. by 1.5-in. was clad in 1/4 inch of stainless steel with a plasma sprayed 5-mil deposit of tungsten forming a diffusion barrier between the uranium and the stainless steel. This test unit was placed in a furnace operating between 1900°F and 1950°F for 2-1/2 hours and, after cooling, was examined to determine the effectiveness of the barrier. No evidence of low melting alloy could be found. It is believed that a 5-mil tungsten barrier will effectively

prevent diffusion alloying between uranium and stainless steel up to at least the melting temperature of uranium, i.e., 2071°F . The quarter-scale model of the demonstration cask,⁸ built with a tungsten diffusion barrier, was also fire tested without evidence of low melting alloy formation. The effectiveness of this procedure was demonstrated under actual fire conditions in the testing of the demonstration cask.

Grain Randomization Development

The fundamental properties of uranium, relative to thermal expansion, and the behavior of the basic metal crystal (allotropy) differ from most structural materials, and many of the difficulties experienced by others when using this metal have been due to a general lack of understanding of these properties. These properties have been thoroughly investigated and documented; however, they are briefly reviewed in the following paragraphs to help explain some of the special fabrication procedures which were developed to produce laminated uranium metal casks. If additional information is desired, several references,^{9,10,11} which are the sources of much of the information presented in this section, are recommended.

Uranium metal has three allotropic phases, generally referred to as alpha, beta, and gamma. The alpha phase is orthorhombic with a density of 19.05 g/cc and is the only phase existing at any temperature less than about 1225°F . The beta phase is tetragonal with a density of 18.10 g/cc and exists between 1225°F and 1422°F . The gamma phase is body-centered cubic with a density of 17.90 g/cc and exists from 1422°F to the melting point of uranium, 2071°F . The density figures given above are typical of the values determined by different investigators. The exact values are difficult to determine experimentally due to the effect of impurities which always exist in test samples. The phase transformation temperatures are also approximate since they depend upon whether the material is being heated or cooled and also upon the normal variance among investigators. The key point as far as laminated cask technology is concerned is the fact that unalloyed uranium undergoes a complete phase change (alpha to beta) when heated into the beta range, i.e., higher than 1225°F , and then reverts completely back to the alpha phase when cooled back to less than 1225°F . Practically speaking, it is impossible for any beta-phase structure to exist in unalloyed uranium at normal operating temperatures. Since the alpha phase is the only allotropic form in which uranium can exist in any uranium structure when the temperature is less than 1225°F , the thermal expansion properties of beta and gamma uranium are not considered in this review.

The coefficient of thermal expansion of uranium is not abnormally large compared to many structural materials; however, the unique property of uranium in this respect is the high degree of anisotropism exhibited in the alpha phase. An anisotropic material is one which exhibits different properties when tested along the different crystal axes. For alpha uranium at room temperatures, the coefficients of thermal expansion are approximately $+12$ and $+13(10^{-6}/^{\circ}\text{F})$ for the "A" and "C" axes, respectively. However, the corresponding value for the "B" axis is $-1(10^{-6}/^{\circ}\text{F})$. Since the values of these coefficients increase appreciably as the temperature

is increased, the actual values for the temperature range being considered should be determined when any thermal expansion calculations are being made. In cases where random grain orientation exists, the actual coefficient of expansion is the same in all directions and is the arithmetic average of the three basic values. In cases where the metal has been rolled in the alpha phase without subsequent heat treatment, some degree of preferred orientation of the metal crystals will exist. The degree of preferred orientation will depend primarily upon the extent of working, e.g., the percent reduction incurred in rolling a billet into plate or an ingot into a slug and to a lesser extent the temperature at which the metal was worked. Generally speaking, uranium in the as-reduced or derby form - or in the form of a casting, will not exhibit any significant amount of preferred orientation and its coefficient of expansion can be specified. The coefficient of thermal expansion for cast uranium, or alpha-worked uranium which has been heat treated to obtain random grain orientation, is comparable to that of normal structural materials. Table 1 lists handbook thermal expansion coefficients for several structural materials and indicates the relationship between uranium and these materials.

Table 1

THERMAL EXPANSION COEFFICIENTS FOR VARIOUS MATERIALS

Material	Thermal Expansion Coefficient, $\times(10^{-6}/^{\circ}\text{F})$
Aluminum	13.3
Copper	10.0
Stainless steel (300 class)	10.0
Stainless steel (400 class)	6.1
Steel	6.7
Monel	7.8
Uranium (random grained)	8.0

In the case of rolled plate which has not undergone a grain randomization treatment, the actual coefficient of thermal expansion will differ appreciably if measured parallel to or perpendicular to the direction of rolling. The exact value of the coefficients in the different directions depends upon the degree of preferred grain orientation and metal texture which exists in the rolled plate.

Due to the anisotropy of the basic uranium crystal, all uranium is subject to physical change by a phenomenon known as thermal cycling growth. Several theories have been developed to explain the mechanism of this phenomenon, and the thermal ratcheting explanation as presented by Holden¹¹ appears reasonable. However, knowledge of the exact mechanism of thermal cycling growth is not required to understand its effect on uranium metal or the treatment required to minimize these effects.

If uranium metal having random grain orientation is subjected to repeated thermal cycling, there is no dimensional change. However, there is some degree of surface roughening which can be attributed to the thermal cycling growth phenomenon. If uranium metal having some degree of preferred grain orientation is subjected to repeated thermal cycling, actual growth in a direction parallel to the preferred grain orientation occurs. The growth rate in microinches per inch-cycle for any specific thermal cycle depends primarily on the degree of preferred orientation. The actual growth coefficient is also dependent on many other factors as indicated below:

1. For the same degree of preferred orientation, the growth coefficient increases as the grain size becomes smaller. This presents a dilemma since for most properties a fine grained structure is preferred.
2. Slow heating coupled with rapid cooling tends to increase the growth coefficient.
3. For the same maximum temperature, a greater cycling range increases the growth coefficient.
4. Holding the sample at the upper temperature for periods up to 30 minutes increases the growth coefficient. Holding periods in excess of 30 minutes have little effect.
5. For equal thermal cycle ranges, the growth coefficient is greater when the maximum temperature is higher.
6. There is no growth unless the upper cycle temperature exceeds approximately 650°F.

The growth coefficients for different samples subjected to the same thermal cycling can vary widely depending on the chemical composition and the prior history of the sample. For this reason, test data are required in order to make a quantitative statement regarding the thermal cycling growth characteristics of any specific uranium item.

There has been a strong tendency in the past to equate thermal cycling growth with radiation induced growth. Although these characteristics are similar in many respects, the growth mechanism is different and certain factors have totally different effects on the subject material. Both mechanisms depend upon the anisotropy of uranium, however, and in both cases growth in any particular direction can only occur when preferred grain orientation exists. It follows, then, that procedures which have been developed to obtain random grain orientation in order to control radiation growth are equally effective in controlling thermal cycling growth. A considerable amount of investigative work concerning the effect of different beta heat treating procedures to obtain random grain orientation in uranium metal, which has been performed by others, was considered in the development of the grain randomizing procedures used for Paducah cask fabrication.

The development effort at Paducah relative to the thermal cycling growth of uranium was not intended to develop basic information but rather to prove the effectiveness of the heat treatment being used. Early work on as-received 1/2-in. uranium plate produced at the Oak Ridge Y-12 rolling facility indicated a growth coefficient of about 85 microinches per inch-cycle when samples were cycled between 150° and 700°F. This coefficient increased to 540 microinches per inch-cycle when the samples were cycled between 150° and 950°F (figure 2). In all cases the growth was parallel to the direction of rolling. A sample checked perpendicular to the direction of rolling showed no measurable growth.

Based on information available in the literature and on experience obtained previously by the Mallinckrodt Chemical Company, Weldon Spring Plant, a tentative beta heat treating procedure was established. A sample plate was subjected to this procedure and samples taken from the test plate showed no measurable growth after 100 cycles from 150° to 700°F. The details of this procedure are given in the previously issued report on the prototype ⁶⁰Co cask.⁶ Basically, the procedure consisted of holding the 1/2-in.-thick uranium plate at 1350°F in a molten salt bath for 30 minutes followed by an oil quench. The highly oriented alpha grain structure which exists in rolled uranium plate is transformed to the beta phase when the metal is held for a relatively short time above 1225°F. When the plate is cooled through the beta to alpha transition temperature, the alpha crystals reform in a random pattern, and subsequent thermal cycling growth is essentially eliminated.

Before building a second prototype, large isotope shipping cask, consideration was given to using a beta heat treating procedure which would eliminate the salt bath procedure. It was reasoned that a suitable procedure might consist of beta heat treating a finished shield after all the laminations had been formed and welded around the inner cladding. This would also stress relieve the individual plates and might result in tighter fitting laminations. A shield heat treated in this manner would cool slowly and have large grains. Although a fine grained structure is usually desired in structural materials, the geometry of a cask is such that a slight loss in strength and ductility is not a factor in the finished product.

To determine the effectiveness of this type procedure, a test was conducted using uranium plate which was typical of that used to produce laminated uranium casks. For this test, seven plates, each measuring 1/2-in. by 6-in. by 12-in., were placed on top of each other to simulate a section of a laminated shield 3-1/2-in. thick. The assembly was canned in stainless steel, heated in a resistance furnace, and air cooled. The 12-in. dimension of each of the test plates was parallel to the direction of rolling when the plate was produced. Thermocouples were inserted into holes drilled into the center and outer plate. The test unit was heated until the temperature of the center plate reached 1350°F and then held at that temperature for one hour. The unit was then air cooled, the stainless steel was stripped from the assembly, and 10-in.-long test specimens were cut from a center and an edge plate. Both samples were cycled 30 times from 100° to 750°F with no detectable change in sample length. They were then cycled 74 more times from

ORIGINAL

AFTER 992 CYCLES

URANIUM GROWTH TEST

Figure 2

200° to 1000°F. Both samples were slightly distorted which prevented precise measurement; however, it was believed that if any growth had occurred, it was less than one microinch per inch-cycle. In a previous test, identical to the test described above, the time at 1350°F was limited to a few minutes, and the thermal cycling growth coefficient was about 25 percent of that found in similar as-received plates. On this basis, it was concluded that assemblies of 1/2-in. uranium plate can be made immune to the harmful effects of thermal cycling growth by holding the assemblies at 1350°F for one hour followed by slow cooling in air.

Uranium Welding Development

Welding of uranium is a key factor in the production of laminated uranium casks. Welds must be free of voids and/or oxide inclusions in order to exhibit the same radiation attenuation as the base metal. Welds must have sufficient strength and ductility to draw and hold the individual laminations tightly together so that the finished cask will have the required structural strength and heat transfer capability. The welding procedure must not be so difficult or tedious that welding costs are a major fabrication cost factor. In addition, the procedure should not present an undue health hazard to personnel doing the welding. Welding procedures have been developed which meet all of these criteria.¹²

The property of uranium which makes it difficult to obtain sound welds is its chemical reactivity at welding temperature, specifically, its high affinity for atmospheric oxygen. An absolute absence of air in contact with hot uranium is necessary if 100 percent sound welds are to be obtained. The uranium welding method used to produce the demonstration cask was the tungsten inert gas procedure which is generally referred to as TIG welding. In this procedure, the parts being welded are melted by an electric arc between a tungsten electrode and the work piece. The immediate area is flooded by a stream of inert gas, such as helium or argon, to prevent oxidation, and filler rods are melted into the molten pool to provide the weld metal. This procedure is generally adequate to weld such reactive metals as aluminum; however, due to its very high reactivity, uranium welds made in this manner tend to contain oxide inclusions which reduce both weld strength and radiation shielding ability.

To produce the uranium welds on the demonstration cask, the basic TIG procedure was modified by silver plating the filler rods. Silver is completely insoluble in molten uranium and forms an impervious film on the molten surface. The silver coating on the filler rod prevents oxide formation on the rod until it is melted during welding. The silver from the rod then floats on the surface of the molten uranium and prevents the formation of insoluble material which would remain in the weld and impair its quality. The silver continues to protect the weld metal during cooling and eliminates the need for chemical or major physical treatment of the weld between passes. Elimination of cleaning between passes increases the overall welding speed by a factor of 5 to 10. Welds produced by this procedure are normally free of voids and/or inclusions and have chemical and physical properties comparable to those of cast or beta heat treated uranium.¹²

Silver is applied to the individual filler rods by electroplating using a standard silver plating procedure. The required amount of silver plate varies with the depth of the weld being produced and ranges from 1/10 to 1 mil in thickness. Since such a small amount of silver is required, the economic factors related to increased welding speed and weld quality far overshadow the cost of silver plating the filler rod. The exact procedures used to produce the plated rod, as well as the detailed welding procedure and the economic factors related to this procedure, are discussed in the report on fabrication of the demonstration cask.

Welding uranium by this method, like many industrial operations, is potentially hazardous to the personnel involved unless specified procedures are followed. In this case, the welding is done in a ventilated booth to prevent general shop contamination and respiratory protection is required for personnel during the actual welding. The general subject of safety for shop personnel is also thoroughly reviewed in the report on fabrication of the demonstration cask.

Chemical Activity Control Development

The property of uranium responsible for many of the difficulties associated with earlier unsuccessful attempts to use this material for radiation shielding is its chemical reactivity. As indicated in a preceding section, the difficulties encountered in welding uranium are due primarily to its chemical affinity for atmospheric oxygen, and the development of a successful welding procedure was based on a technique to prevent this reaction.

Control of chemical reaction between uranium and its environment is a key factor in the successful use of this metal in radiation shielding applications. During cask fabrication, the major health hazard to personnel working with uranium is not exposure to radiation from the uranium but from ingestion or inhalation of uranium oxide formed during such operations as heating, grinding, welding, etc. Procedures developed to control this hazard involved such techniques as covering uranium metal with oil or coolant during machining or hot forming, respiratory protection for personnel doing welding and grinding, and performing certain operations within a booth designed to exhaust fumes from the fabrication shop. For example, in forming the laminations it is necessary to keep the uranium plate at a temperature in excess of 400°F (warmer than the brittle transition temperature) to keep it from cracking. Coating of uranium with a silicone fluid was found to reduce oxidation of uranium in air at 600°F sufficiently to permit utilization of open-air furnaces for this heating operation.

Uranium shields which dissipate large amounts of decay heat must have low overall thermal resistance, and the presence of oxide films is very detrimental in this respect. This is particularly true of a laminated structure where the heat generated within the cask cavity must pass through many interfaces. Procedures which minimize oxide film formation, both during fabrication and also while the unit is in service,

are an important part of the technology required to use laminated uranium in radioactive material shipping casks. The procedures used to minimize oxide formation on uranium surfaces during cask fabrication are discussed in detail in the report covering cask fabrication.

The key to preventing chemical reaction after a unit has been placed in service is cladding integrity. It is imperative that the cladding be absolutely free of any breaks or pin holes if satisfactory service is to be obtained. The proper design of welds along with adequate inspection and test procedures are mandatory in order to prevent cladding failure. These items are discussed in detail in a later section of this report. A great deal of investigative work has been done by others relative to the use of clad uranium metal as fuel in water reactors.⁹ Much of this work relative to water induced corrosion is applicable to uranium shipping casks. The obvious reactions are those between uranium and oxygen in air, moisture in air, or water to form uranium oxide. The corrosion products in this type of reaction are generally loose and fluffy and quickly fill any void space within the cladding of a shipping cask. This condition tends to halt the reaction since further contact with water is prevented by the presence of the previously formed reaction products.

The chemical reaction between uranium metal and hydrogen to form uranium hydride is not self limiting, however, and can cause failure of cask cladding, although not a catastrophic-type failure of a cask. Hydrogen which is formed when uranium reacts with water migrates through extremely small passages to locations where the most favorable conditions for its reaction with uranium exist. The optimum temperature for this reaction is about 350°F which is typical of that found within shipping casks. Uranium hydride formed by the reaction can split cladding or cause bumps which could prevent the removal of containers placed within the cask cavity. An insidious characteristic of a deposit of uranium hydride is that it will decompose upon an increase in temperature leaving a spongy residue and free hydrogen. The liberated hydrogen will migrate to a more optimum location within the cask and react with more uranium to form additional deposits of uranium hydride. The cycle can continue indefinitely following the admission of a relatively small amount of water through the cask cladding. Uranium hydride is also unstable in the presence of air or moisture, forming uranium oxide and liberating hydrogen.

PROGRAM OBJECTIVES

The primary objective of the demonstration laminated uranium fuel element shipping cask project was to demonstrate, by testing a full scale unit, that a uranium cask could be designed and fabricated which would meet all of the performance and hypothetical accident criteria as specified in the applicable regulations. In developing the detailed program to meet this overall objective, seven basic objectives were outlined as a guide to the uranium cask fabrication technology program. These basic objectives were as follows:

1. Demonstrate actual performance, fabrication procedures, and costs of a prototype demonstration fuel element shipping cask made of laminated uranium metal. One of the objectives of the project was to actually demonstrate that a laminated uranium cask could be designed and fabricated which would satisfactorily perform as a fuel element shipping cask. Practical considerations suggested that it should also be demonstrated that the cask could be immersed into a tank of cold water (while carrying a simulated spent fuel decay heat load) without experiencing any structural failures and that the lid could be readily inserted into the cavity while the cask was submerged in a tank of water. Collection of accurate cost data was also desired relative to the fabrication of the demonstration unit. The data developed in this program should permit a valid comparison between casks built using this method of construction and casks built by other methods using uranium (e.g., castings) or other materials (e.g., lead). Information derived from objective 4 below is intimately associated with this part of the program.
2. Demonstrate structural integrity under accident conditions as specified in the applicable regulations. The hypothetical accident conditions involve the ability of a designated unit to survive a free fall from 30 feet onto an unyielding surface; a free fall from a height of 40 inches onto an unyielding piston; and an exposure to a 1475°F fire for 30 minutes. In most cask designs, destructive tests of the full scale units would be prohibitively expensive and, as a result, engineering evaluations have been used as a substitute to assess actual performance under the specified conditions. In some instances, model testing has been employed to further substantiate the engineering evaluations. In this program, actual test data would be obtained on the demonstration uranium cask, thereby providing reliable information on cask integrity.
3. Demonstrate compliance to performance criteria as specified in the regulations as well as the design criteria developed specifically for this cask. Included in this objective are such important factors as demonstrating the ability of the

cask to (1) dissipate the heat load from the specified number and type of spent fuel elements, (2) provide the required degree of radiation attenuation, and (3) satisfactorily withstand exposure to a -40°F ambient temperature.

4. Develop detailed fabrication cost data to accurately predict the costs of building any size laminated uranium cask. It has been noted above that the actual total fabrication cost would be determined. By carefully analyzing all cost factors and closely observing the actual fabrication operation, basic engineering cost factors would be developed which should permit any fabricator to determine his costs to build similar casks. Such factors as man-hours per foot of weld, man-hours per square foot of uranium plate to form laminations, inspector man-hours per foot of weld to dye check welds, time required to heat treat and stress relieve shielding assemblies, etc., would also be developed. Basically, the entire cost of fabricating the demonstration cask would be broken down into all of its component parts.
5. Develop engineering criteria which can be utilized to design laminated uranium casks to meet desired performance criteria. Data obtained during the testing sequence on such factors as heat transfer capability, stresses and g forces developed under impact conditions, rate of temperature rise within the cask during fire exposure, etc., would be useful in future designs of laminated uranium casks for radioactive material shipments. The data generated in this program will be a major contribution to the technical knowledge in the field of engineering evaluation of uranium cask performance and can be used in future evaluations of uranium casks by personnel concerned with safety and licensing of casks for shipment of radioactive materials.
6. Provide test data for use by others conducting related development programs or preparing engineering evaluations of similarly designed casks. Various phases of the many problems involved in transporting radioactive materials are being studied at many sites. In many cases, data are only available over very limited ranges from model testing, and considerable extrapolation must be employed. The data collected in this program will permit extension of the areas of actual test data and minimize the degree of extrapolation that must currently be employed by cask design engineers. In addition to the uranium data obtained, valuable data will also be obtained on the stainless steel components of the cask regarding shock absorption capability, weld integrity under all conditions, etc. Modified lid

closure and shock absorbing fin designs will also be tested and should provide valuable data over a range of stresses considerably above those determined in most programs.

7. Document performance of the demonstration cask in a comprehensive series of reports. Separate technical reports would be prepared covering the following various phases of the project:
 - (1) General information relative to the use of uranium as a cask component and the design of the demonstration cask.
 - (2) Fabrication of the demonstration cask and the economic factors relative to the fabrication and use of laminated uranium metal shipping casks.
 - (3) Testing of the demonstration cask.

PROGRAM CRITERIA

SELECTION OF BASIC SHAPE AND SIZE

It was indicated in the Schedule 189a¹ that the overall project planning would be coordinated with Chemical Technology Division personnel at ORNL. This was done to utilize their shipping cask background and experience since the bulk of Paducah experience was related to the technology of uranium fabrication rather than the broad aspects of shipping cask design. Accordingly, the initial program planning meeting was held at ORNL on July 27, 1966, with personnel from the ORNL and the Paducah Gaseous Diffusion Plant in attendance. Basic cask philosophy developed at the meeting is outlined below:

1. The actual demonstration cask should incorporate features applicable to casks weighing from 50 to 75 tons.
2. The demonstration cask should weigh from 12 to 15 tons. A cask of this size would permit valid extrapolation of the test results and be large enough to demonstrate fabrication techniques applicable to full-size casks.
3. A cask having the basic shape of a right circular cylinder with an end closure and a large L/D ratio was selected as being typical of casks which will be used to ship spent fuel from future power reactors. The demonstration cask should have this general shape.
4. The length of the cask should be sufficient to demonstrate fabrication techniques which would be applicable to full-length casks.
5. The demonstration cask should be a practical design; however, since the prime program objective was to demonstrate performance of a uranium cask, it would not be necessary to design, build, and test features which would be common to any cask. Included in this category are such features as supplemental heat removal systems, cask tie-down systems, pressure relief systems, etc.
6. The demonstration cask should be designed to the same criteria as an existing lead cask to provide a good comparison between the two methods of fabrication. The selection of the existing cask should be made to provide the maximum of comparative information and effectively demonstrate the potential advantages inherent to a uranium shielded cask.
7. The latest draft of 10 CFR 71¹³ and other applicable regulations would be used to determine general design and performance criteria. Design standards being developed at ORNL would be used when available and applicable.

SELECTION OF INDIAN POINT CORE A CASK FOR COMPARISON

Since it was desired to design the demonstration cask to the same criteria as had been used for an existing lead cask, it was suggested by ORNL personnel that either the cask designed for Dresden Type I or Indian Point Core A spent fuel assemblies would be suitable for comparison, and a review of these casks was made. A comprehensive description of these casks and their intended usage was obtained from the safety evaluation reports of the USAEC Division of Licensing and Regulation, Docket Nos. 70-729 and 70-951, respectively. Based upon information from these documents, the information shown in table 2 was developed. Since an inherent advantage of uranium casks is a high load factor, i.e., a large quantity of radioactive material shipped per total weight of the package, it was decided that a demonstration cask should have the ability to carry twice as many fuel elements as the lead cask to which it would be compared. As indicated in table 2, a uranium cask could be built which would carry twice the load of the Indian Point cask with slightly less weight of shielding material. For the Dresden cask, the advantage of uranium over lead in this respect was not quite so favorable.

The weight advantage and the fact that a quarter-scale model of the Indian Point cask had been built and tested were the major factors in the decision to compare the demonstration cask to the Indian Point Core A cask. It was subsequently decided that the demonstration cask would be designed to be a prototype of a cask which could carry four Indian Point Core A spent fuel assemblies except that its length would be shortened to reduce its weight to about 24,000 pounds in order to minimize the overall project cost. The design basis for radiation attenuation and heat dissipation would be the same as those of the Indian Point cask to effect the best possible comparison between the two methods of cask fabrication.

BASIC CRITERIA

The original criteria used to design the demonstration cask were developed from several sources which included 10 CFR 71, Docket No. 70-951, table 2, and factors such as the self imposed weight limit of 24,000 pounds. The basic criteria included the following:

1. The cask would have a cylindrical cavity measuring 17 inches in diameter by approximately 54 inches deep. The exact depth would be determined by the estimated cask weight. Based upon the dimensions of Indian Point Core A fuel elements, as given in Docket No. 70-951, four elements could be placed in a cavity 17 inches in diameter.
2. The shielding would be produced from laminations of 1/2-in.-thick rolled uranium plate. This is the same procedure used to produce large radioisotope shipping casks.
3. The uranium would be depleted in ^{235}U and be relatively pure. The rolled plate would be produced from billets cast from

Table 2

COMPARISON OF TWO EXISTING FUEL ELEMENT SHIPPING CASKS

Factor Compared	Cask No. 1	Cask No. 2
Used by	Dresden	Indian Point Core A
Date approved	2-13-63	1-21-66
Designation	GE 13 x 129	GE 15 x 140
Builder	Knapp Mills	Knapp Mills
Applicable standards	10 CFR 72, Sept. '61	10 CFR 71, Dec. '65
Cavity size, in.	13 x 129	15 x 140
Outside diameter, in.	32	33-3/8
Length, in.	147	160
Weight, lbs	---	55,000
Lead thickness, in.	8-3/8	8-1/2
Steel thickness, in.	1/2 inner, 3/4 outer	3/4 inner, 1-1/4 outer
Design capacity	4 assemblies	2 assemblies
Assembly size, in. ²	4.4	5.7
Assembly active length, in.	111.75	100.725
Basket size, in.	12.96 x 125	15 x 138-7/8
Basket, center length, in.	---	108
Design heat load, Btu/hr/lin.ft of fuel	---	4,000
Nominal decay heat load, Btu/hr/lin.ft of fuel	534	1,290
Nominal decay heat load, Btu/hr/lin.ft of fuel with 2X load ^a	4,272	5,160
Inside diameter needed for 2X load, in.	18 +	16.1
Inside diameter assumed for 2X cask, in.	19	17
Shielding assumed for 2X cask, in.	1/4	1/4
Uranium thickness for 2X with same degree of attenuation, in. ^b	5-1/4	5-5/8
Ratio of weight U for 2X cask per weight Pb for 1X cask	1.17	0.985
Outside diameter for 2X uranium cask, in.	30-1/2	28-7/8
Ratio of weight Pb for 2X cask to weight Pb for 1X	1.37	1.16

^a 2X Load means twice the number of elements for which the cask was designed.

^b Assumes a U to Pb ratio of 0.57, 1/4-in. additional U for the 2X load, and uranium equivalent to the steel in the lead cask.

charges containing standard production uranium derbies with no recycle material. The derbies normally have about 150 to 200 ppm of total impurities.

4. The cask would be shipped in the horizontal attitude. Even though the normal shipping position is horizontal, such a cask would be loaded in a vertical position and its heat dissipation surfaces must function adequately in this attitude.
5. The cask would have an end closure with recessed bolts. The angling set screw closure bolting system used on the Indian Point Core A cask would be given serious consideration.
6. The cask would contain pressurized water to conduct heat from the fuel to the cask inside wall.
7. The cask would be internally and externally clad with stainless steel. The cladding prevents surface oxidation during service and subsequent decontamination operations, stops alpha and beta radiation which emanates from the surface of the uranium, and prevents the possible spread of radioactive material which could be rubbed off a bare uranium surface.
8. Shielding would be 5-3/4 inches of uranium or its equivalent. This value was originally determined by calculating the thickness of uranium equivalent to the lead and steel in the Indian Point cask and adding 1/4 inch of uranium to compensate for the increased radiation resulting from two additional spent fuel elements. A more rigorous calculation was subsequently made which predicted the radiation level three feet from the cask surface would be 1.13 mr/hr. However, this calculation neglected any contribution from the uranium shielding. The latter, rigorous calculation and the subsequent radiation attenuation test indicated the 5-3/4-inch criteria to be slightly more than needed.
9. The design decay heat load would be 8,000 Btu/hr/ft of cavity length. This figure is somewhat greater than would be expected from four spent Core A elements, but it was used to maintain the same design basis as was used for the lead cask.¹⁴
10. The cask would be designed to operate in bright sun with an ambient temperature of 130°F.
11. There would be a vent penetration in the lid and a drain penetration in the base.

12. The cask would actually be subjected to the first three hypothetical accidents outlined in 10 CFR 71 and would be tested to demonstrate its ability to meet the normal transit conditions and the design heat dissipation and radiation attenuation capabilities.
13. The cask would be instrumented during the testing program to demonstrate compliance with design and to obtain quantitative data; however, no shielding penetrations would be made solely for instrumentation purposes.
14. A thermal diffusion barrier would be placed between all uranium and stainless steel interfaces.
15. The rolled uranium used in the cask would be given a grain randomizing heat treatment to stabilize the thermal coefficient of expansion and eliminate the possibility of thermal cycling growth. It was realized that the operating temperatures of a full-size cask built to the same design as the demonstration cask would not be hot enough to cause thermal cycling growth; however, a grain randomizing treatment was specified to demonstrate a fabrication step which would probably be required for future casks which may operate with a dry cavity.

DESIGN

CONCEPTUAL DESIGN

The design goal of incorporating all program criteria into a functional cask required the generation of previously nonexistent information concerning unique aspects of fabrication and operation of a laminated uranium shielded cask. A preliminary investigation of factors related to fabrication of the cask was undertaken to establish the optimum thickness of the uranium plate.

In an attempt to provide minimum thermal resistance in the laminated wall, two attempts were made to form cylinders from 3/4-inch thick uranium plate which is the maximum thickness our rolls will handle. These tests were unsatisfactory due to surface roughness produced on the plates. More than 90 percent of the exterior, or tensile surface of the cylinder, was covered with small conical shaped pits. The depth of these pits, although not measured, appeared to be between 0.025 inch and 0.050 inch. It was felt that the net effect on heat transfer would be negative when 3/4-inch thick plates with rough surfaces were compared to smooth 1/2-inch plates. Paducah Plant experience with rolling 1/2-inch plate⁶ had shown little surface roughening so 1/2-inch plate was specified for the demonstration cask.

A close tolerance on plate thickness was called for since each lamination on this cask was to be formed from two cylindrical sections. Experience has shown that when circumferential joints of this type are required on the laminations the plate thickness tolerance should be the minimum possible value. The thickness should not vary more 0.010 inch from plate to plate.

Investigation of factors related to the operation of a laminated uranium cask such as optimum heat transfer and impact resistance indicated the need for development of information concerning the response of a laminated uranium structure to mechanical and thermal loading.

The required information was obtained through testing of a specially constructed laminated uranium-stainless steel cylinder, a radioisotope shipping cask of laminated uranium-stainless steel construction,⁶ and a quarter scale model of the demonstration cask⁸ on which design ideas were tried prior to their incorporation into the demonstration cask.

Response of a Laminated Uranium Cylinder to Mechanical Shock

The low values of energy absorption (10 to 20 ft-lbs at room temperature²²) reported for impact testing of uranium Charpy specimens indicated that the cask would have to include features specifically designed to prevent high energy absorption in small volumes of uranium and corresponding stress concentrations. In an effort to determine the amount of energy absorbed per unit volume of uranium for the loading rates resulting from a 30-foot free fall, an open ended, laminated uranium cylinder clad in stainless steel was fabricated and drop tested. The test cylinder, shown in figure 3, consisted of three laminations, each 1/2 inch thick, clad in stainless steel.

OPEN ENDED LAMINATED CYLINDER
USED FOR IMPACT TESTING

Figure 3

The first test of the cylinder was a corner drop in which the primary energy absorption mechanism was plastic crushing. Assuming all of the energy of the 30-foot free fall was absorbed in the crushed volume shown in figure 4, the uranium absorbed approximately 12,500 inch-pounds of energy per cubic inch of deformed volume.

Following the corner drop, the cylinder was dropped with its longitudinal axis parallel to the impact pad surface. Since the cylinder was open ended, there was a large amount of circular bending as shown in figure 5. Sectioning of the cask indicated there was no radial cracking of the shield such as might be expected with large deformations of this type. Apparently the reason for this was that the shell was laminated rather than solid. A solid shell would develop a very large tensile stress on the inside surface of the thick walled cylinder when subjected to a deformation of this magnitude. The laminated shell, however, apparently acts like a series of membranes, each generating relatively small surface stresses when subjected to a large deformation. Sections cut from the test cylinder are shown in figure 6.

These two drop tests led to the conclusion that a laminated shell possessed the properties necessary for its use as a structural member of the cask. The laminated cylinder responds like a monolithic cylinder when subjected to end loads and like a series of thin wall shells when subjected to side loads.

Thermal Conductivity Determination

A laminated uranium cask with an inside diameter of 11 inches and an outside diameter of 25-1/2 inches was subjected to a heat transfer test in which 19,520 Btu/hr was generated in the cavity. At the midplane, temperature measurements indicated the thermal conductivity of the laminated structure was 6.5 Btu/hr-ft-°F at an average temperature of 312°F. The test cask was made up of fifteen laminations. Since the proposed thickness of 5.75 inches of shielding on the demonstration cask would require a total of fourteen laminations (including the inner and outer cladding), this value of thermal conductivity was used in the demonstration cask heat transfer calculations.

Thermal Design of the Conceptual Cask

Using the heat load and cask size developed earlier in this report, a preliminary analysis was conducted to determine the number and size of circumferential heat transfer fins necessary to dissipate 8,000 Btu/hr per foot of cavity length from a 30-inch O.D. cylinder as shown in figure 7. One dimensional (radial) heat transfer was assumed so that the calculations could be done on a per foot of cylinder length basis. It was also assumed that the fin efficiency, η , would be 50 percent, the solar input would be 25 percent of the internal heat load, and the average surface temperature used for property determination would be 250°F.

The following data were used to determine the temperature distribution:

Heat load	=	8000 Btu/hr-ft of cavity
I. D.	=	17 inches
O. D.	=	30 inches
Emmissivity ¹⁵	=	0.6
Ambient temperature	=	130°F

**OPEN ENDED LAMINATED CYLINDER
AFTER CORNER DROP**

Figure 4

3173-3 (U)

OPEN ENDED LAMINATED CYLINDER
AFTER SIDE DROP

3173-4 (U)

Figure 5

SECTIONS CUT FROM OPEN ENDED
LAMINATED CYLINDER FOLLOWING TESTING

Figure 6

3163-6 (U)

$$T_s = 250^\circ\text{F}$$

$$\epsilon_s = 0.6$$

-LEGEND-

q = Internal heat load per foot of cavity length.

q_s = Solar heat load.

T_s = Surface temperature.

ϵ_s = Emissivity

NOTE: All section faces are adiabatic surfaces due to the one dimensional heat transfer assumption.

SECTION THROUGH LAMINATED URANIUM CASK WALL

Figure 7

Using figure 8, the radiative heat transfer coefficient, h_r , was found to be 1.12 and the convective heat transfer coefficient, h_c , was found to be 0.92. The total heat transfer coefficient was then calculated by:

$$h_t = (h_r + h_c) = 2.04 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F} \quad (1)$$

For the specified temperatures of 130°F ambient and 250°F surface, the heat flux, q , corresponding to this total heat transfer coefficient is given by:

$$q = 2.04(250-130) = 245 \text{ Btu/hr-ft}^2 \quad (2)$$

The required total area per foot of length to give this flux when the internal load is 8000 Btu/hr-ft²-ft and the solar load is 2000 Btu/hr-ft²-ft is:

$$\frac{8000 + 2000}{245} = 40.8 \text{ ft}^2/\text{ft} \quad (3)$$

Since there is 7.85 ft²/ft of surface area on a 30-inch cylinder, an effective area of 32.95 ft²/ft of fins is needed. The area of a 30-inch I.D., 36-inch O.D. fin is:

$$\frac{2\pi(36^2-30^2)}{(144)(4)} = 4.34 \text{ ft}^2/\text{ft} \quad (4)$$

At 50 percent efficiency, this is equivalent to 2.17 ft²/ft. The total fins per foot required are then:

$$\frac{32.95}{2.17} = 15.2 \text{ fins/ft} \approx 15 \text{ fins/ft} \quad (5)$$

It is impractical to put fifteen fins per foot on a cask of this size so detailed calculations were made using twelve fins per foot which was thought to be the largest practical number. In the detailed calculations, the radiative heat transfer coefficient, h_r , was corrected for the fin geometry shown in figure 9 by:

$$h_r' = (1-F)h_r = 0.336 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F} \quad (6)$$

where F = the view factor = 0.7
 h_r = radiative heat transfer coefficient for a view factor of zero
 h_r' = corrected heat transfer coefficient

The total heat transfer coefficient used in the calculations was then:

$$h_t = (h_r' + h_c) = (0.336 + 0.92) = 1.26 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F} \quad (7)$$

The fin efficiency was calculated for the fin geometry by:¹⁶

CONVECTIVE AND RADIATIVE HEAT TRANSFER
COEFFICIENTS AS A FUNCTION OF TEMPERATURE
DIFFERENCE WHEN AMBIENT TEMPERATURE IS 130°F

Figure 8

FIN SPACING ON DEMONSTRATION CASK

$$\eta = \frac{\text{Tanh}(ML)}{ML} = 74\% \quad (8)$$

where $L = \text{fin length} = 3 \text{ in.} = 0.25 \text{ ft.}$

$$M = \sqrt{\frac{2h_t}{kt}} = 4.24 \text{ ft}^{-1}$$

$$h_t = 1.26 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$$

$$k = \text{thermal conductivity of stainless steel}^{17} = 9 \text{ Btu/hr-ft-}^\circ\text{F}$$

$$t = \text{fin thickness} = 3/16 \text{ in.} = 0.0156 \text{ ft}$$

For the above values, $\eta = 74\%$. This is much greater than the 50 % efficiency assumed and gives an effective fin area of:

$$A_F = \frac{2\pi(36^2-30^2)(12)(0.74)}{(144)(4)} = 38.2 \text{ ft}^2/\text{ft} \quad (9)$$

The cask area less the fin area is:

$$A_c = \pi \left(\frac{30}{12} \right) [1-12(0.0156)] = 6.3 \text{ ft}^2/\text{ft} \quad (10)$$

And the total surface area is:

$$A = 38.2 + 6.3 = 44.5 \text{ ft}^2/\text{ft} \quad (11)$$

Since this is greater than the required total area of 40.8 ft²/ft calculated in equation (3), twelve fins per foot will achieve the desired heat transfer when the fin efficiency is corrected to 74 percent from the originally assumed 50 percent.

The total direct and indirect solar input to the cask was then calculated to be 1776 Btu/hr-ft²-ft by the method proposed by L. B. Shappert.¹⁵

The total surface heat load was then:

$$8000 + 1776 = 9776 \text{ Btu/hr-ft}^2\text{-ft} \quad (12)$$

This resulted in a flux of:

$$\frac{9776}{44.5} = 220 \text{ Btu/hr-ft}^2 \quad (13)$$

and a surface temperature of 240°F from figure 10. This surface temperature agrees closely with the 250°F assumed in determination of h_c and h_r' . The inside wall temperature was calculated by adding the temperature drop through the wall to the surface temperature. The temperature drop in the wall is given by:¹⁶

$$\Delta t = \frac{q' \ln r_2/r_1}{2\pi k} = 111^\circ\text{F} \quad (14)$$

**SURFACE TEMPERATURE AS A FUNCTION
OF SURFACE HEAT FLUX
WHEN THE AMBIENT TEMPERATURE IS 130°F**

Figure 10

$$\begin{aligned}
 \text{where } q' &= 8000 \text{ Btu/hr-ft}^2\text{-ft} \\
 r_2 &= \frac{\text{O.D.}}{2} = 15 \text{ inches} \\
 r_1 &= \frac{\text{I.D.}}{2} = 8.5 \text{ inches} \\
 k &= 6.5 \text{ Btu/hr-ft-}^\circ\text{F}
 \end{aligned}$$

The inside wall temperature corresponding to this temperature drop is:

$$240^\circ\text{F} + 111^\circ\text{F} = 351^\circ\text{F} \quad (15)$$

The saturation pressure for water at 351°F is approximately 136 psia or 121 psig.¹⁸

The calculations above are based on a surface flux of 220 Btu/hr-ft^2 and a radiative and convective heat transfer from a surface with an emissivity of 0.6 to 130°F still air.

Mechanical Design of the Conceptual Cask

The valves selected for use on the cask were ball valves with a Teflon primary seat and a metal secondary seat. A rupture disc was used to protect the cask against damaging pressure buildup.

Since the primary purpose of this program was to develop a technology that would allow fabrication and use of uranium shielded casks, no effort was made to develop optimum valving. However, experience gained during fabrication and testing of the demonstration cask indicates that valve systems on fuel element casks involving external piping and commercially available valves are difficult to protect and service. A more desirable solution to the problem would be to eliminate all external piping and mount a specially constructed valve in a protected area on the surface of the cask. The valve inlet line should be completely contained in the shielding. The primary seal should be metal and secondary seals should be provided to contain leakage past the stem or the seat.

The closure on the conceptual cask was designed to prevent radiation streaming through the crack between the plug and the cask. Since the penetrations into the top and bottom of the cask reduce the effective shielding thickness in these areas, additional material was provided to maintain a minimum of 5.75 inches of uranium or equivalent between any point inside the cask and any point outside the cask.

The conceptual design based on the foregoing assumptions and calculations is shown in figure 11.

CONCEPTUAL DESIGN OF THE DEMONSTRATION FUEL ELEMENT SHIPPING CASK

Figure 11

PHASE I DESIGN

Development of Final Design

Discussions of the general features of the cask were held with ORNL and AEC personnel for the purpose of comparing the conceptual design to current good cask design practices. Several constructive comments resulted from this meeting. They were: (1) the male lifting trunnions should be replaced by female trunnions to reduce the chance of cladding damage at impact, (2) the silicone rubber O-ring gasket should be replaced with a higher service temperature rated elastomer gasket, (3) the valve on the lid should have more protection, and (4) a small taper should be provided on the lid, rather than the straight cylindrical section as shown, to facilitate lid installation and removal.

Two of these comments affect the performance of the cask under impact conditions; so a revised drawing of the cask was started prior to any attempt at a comprehensive study of the mechanical behavior of the cask at impact. A decision was reached to place fins on the closure end of the revised cask in an effort to reduce the shock loads on the cask and contents. This came about after extrapolating the test results of the laminated cylinder mentioned earlier to the conceptual cask. The extrapolated data indicated that accelerations in the range of 1000 g to 1500 g could be expected if the solid uranium section of the cask made initial contact with the impact pad at the end of a 30-foot free fall.

The revised drawing, called the Phase I design, included impact protection fins, provision for shipment in the horizontal position, and changes to satisfy comments on the conceptual design. The Phase I design, shown in figure 12, was 6.5 inches longer outside and 2.25 inches shorter inside than the conceptual cask. The silicone rubber O-ring gasket was replaced with two gaskets: one of hollow nickel and one of fluorocarbon rubber (DuPont Viton). The nickel O-ring will withstand continuous service at more than 1000°F and will maintain a seal during short-term exposure to much higher temperatures.¹⁵ The Viton O-ring has a continuous service temperature range from -20°F to 400°F,¹⁹ and it will withstand short-term exposure to temperatures as high as 750°F. In addition, Viton is less susceptible to radiation damage than most elastomeric gasket materials.

The "egg crate" structure on the base of the cask was extended to provide additional valve protection and, for the same reason, the valve at the top of the cask was recessed into a finned structure.

All welds on the Phase I design are located in areas where there is no large deformation during impact. In addition, all cladding welds on the outside are reinforced. The longitudinal seam on the cladding is reinforced by the fins, the base-plate weld is reinforced by "egg crate," and the cladding-to-stud ring weld is protected by a 1/2-inch plate welded over it.

A shipping pallet, or skid, was incorporated in the Phase I design to facilitate horizontal shipment of the cask. The rotation of the cask

PHASE I DESIGN OF THE DEMONSTRATION FUEL ELEMENT SHIPPING CASK

Figure 12

from vertical to horizontal was accomplished by placing two trunnions, added to the base of the cask, over pins on the skid. When in the horizontal position, the cask was supported by the upper and lower trunnions.

Quarter Scale Model

A decision to proceed with detailed shop drawings was reached at the conclusion of the Phase I design work. At this time, a quarter scale model was built to verify that the design satisfied program criteria and to test the support equipment that was to be used in the demonstration cask tests.

The quarter scale model was built without impact fins, heat transfer fins, lifting trunnions or valves in order to keep fabrication costs to a minimum. The uranium sections of the model were machined from castings to simplify construction, and penetrations were provided through the top and bottom of the model to facilitate internal instrumentation. The model as originally constructed is shown in figure 13.

Closure integrity was considered to be vitally important during the design work. Since the Phase I closure was an adaptation of lead cask technology untried on uranium structures, the initial model test was a 40-inch "shake-down" drop with the model impacting on the closure edge. The test indicated the angling set screw closure was inadequate for this type of a cask. The probable differences between this test and successful lead cask tests using the same closure are: (1) this cask has a smaller length to diameter ratio than lead casks with similar closures, such as the Indian Point cask, and (2) the shock loads in a rigid uranium structure are higher than in a lead structure. The smaller length to diameter ratio results in greater departure from vertical on closure corner drops and, therefore, higher moment generation in the lid. The failure of the closure to perform as desired apparently was caused by two factors that were underestimated during the design. These were: (1) the effect of the moment generated on the screws and (2) the effect of the difference in hardness between the screws and the stainless steel face of the lid.

The shock loads in this test were low since the drop height was small. The average acceleration during impact in this drop was estimated to be 320 g. The expected average in the 30-foot fall of this particular model was in the 1500 g to 2000 g range. The angled set screw closure could not be modified or improved enough to withstand loads in this range when the moments and stress concentrations that had been overlooked were considered. A complete redesign of the closure was considered necessary.

PHASE II DESIGN

Free Fall Analysis

Design work was shifted back to the demonstration cask, and an analysis of impact conditions was undertaken in order to approximate the loads

Figure 13

the closure would experience in the 30-foot free fall. The analysis was based on a rigid shield, sacrificial-external-member concept in which the uranium shield would support fins and other extended stainless steel members that could be destroyed in an accident without compromising the integrity of the cask. The depth of crushing was essentially limited to the depth of the fins by the fin design; so the fin depth is a good approximation of the distance in which the cask stops. The stopping distance and the free fall distance are all that are needed to solve the equations of motion for the average stopping acceleration. This is demonstrated for a 30-foot free fall stopped in four inches by writing the equation relating distance, acceleration, and velocity:²⁰

$$V_{30} = \sqrt{2 as} = \sqrt{60 g} \quad (16)$$

where $s = 30$ feet
 $a = 1 g$

the velocity, V_{30} , is the velocity at the end of the downward acceleration and the beginning of the upward or stopping acceleration. Therefore:

$$V_{30} = \sqrt{2 as} = \sqrt{2/3 a} \quad (17)$$

where $a =$ average stopping acceleration in g's
 $s = 4$ inches $= 1/3$ foot

equating (16) and (17) yields:

$$\sqrt{2/3 a} = \sqrt{60 g} \quad (18)$$

where $a = 90 g$

The time that elapses while the cask is coming to rest is given by:

$$t = \sqrt{2 s/a} = 0.0152 \text{ second} \quad (19)$$

where $s = 4$ inches $= 0.33$ foot
 $a = 90 g = 2880 \text{ ft/sec}^2$

The peak "g" loads on the cask were approximated by assuming the new fin design would be similar to figure 14a. The dynamic load necessary to cause elastic buckling of the fixed end fin was then assumed to be approximately the same as the static load that would buckle a pivot-ended fin as shown in figure 14b. The pivot-end assumption was made in the interest of calculating the smallest load the fin would support under any condition. If the collapse load was larger than the calculated load, it would indicate a greater energy absorbing capability. The case in which the energy absorbing capability was the poorest was calculated since the actual collapse mechanism was unknown due to the complex fin geometry and unknown loading rate and direction.

Fig. 14 a

Actual fin used on cask.

Fig. 14 b

Pivot ended fin
used for peak g load
calculations.

NOTE: p = the buckling load.

IMPACT FINS

For the fins shown in figure 14b, the buckling load, P, is given by:²¹

$$P = \frac{\pi^2 EI}{l^2} = 7.7 \times 10^6 \text{ lbs} \quad (20)$$

where E = Young's Modulus = 30×10^6 psi

l = 4 inches

$$I = \frac{b(h^3)}{12} = \frac{5(1^3)}{12} = 0.417 \text{ in}^4$$

b = 5 inches (base of fin)

h = 1 inch (thickness of fin)

This corresponds to an initial peak of approximately 275 g. Other fins were assumed to come in contact with the pad at regular intervals so that a uniform force of one-fourth the initial force would be maintained on the cask through the 4-inch travel. Calculated peak accelerations at impact were then plotted over average acceleration on an acceleration time graph as shown in figure 15. Comparison of the area under the peak curve to area under the average curve indicates that all of the energy will be absorbed in crushing of the sacrificial members if these members react to impact loading in the manner assumed.

The above calculation of peak g load during impact is an approximation involving many assumptions. Exact analysis is impossible due to the complexity of the geometry and the stress field generated during fin collapse. Since neither analytical nor modeling techniques for predicting the amount of energy that can be absorbed by deformation of various fin configurations has been fully developed,¹⁵ a designer wishing to extend this fin configuration to casks of other sizes and shapes is urged to exercise caution. This fin design functioned as calculated in the 30-foot free fall test of the quarter scale model, but something less than the desired amount of energy absorption by fin deformation resulted from the drop test of the 28,000-pound demonstration cask. The smaller percentage of energy absorption by the fins on the demonstration cask still met the program objectives. However, it does point out that the finned energy absorbing structure should not be scaled from cask to cask on geometrical considerations alone.

Closure Redesign

Examination of cask closure systems currently in use indicated the stud type closure as the most practical replacement for the angled, set screw closure used on the Phase I design. Studs were chosen over bolts or cap screws because of the decontamination problems associated with blind, tapped bolt holes.

The problem of getting the lid into place over a large number of studs was considered in the new design. Clearances between studs and holes and between the lid and cask were maintained at what was thought to be the maximum amounts compatible with other dimensions. Even so, the lid occasionally jammed in the cask while being inserted. The principal problem appeared to have been lack of enough clearance between the

**CALCULATED AVERAGE AND PEAK ACCELERATION CURVES
FOR DEMONSTRATION CASK 30 FOOT FREE FALL TEST**

studs and the holes. The problem could be reduced somewhat by enlarging the holes in the flange or bellling the mouth of the hole. Any changes in clearance between parts, however, would have to be accompanied by additional changes in the closure to prevent damaging lateral motion¹⁵ of the lid relative to the cask at impact.

The quarter scale model was rebuilt to incorporate these design changes. Subsequent testing of the model indicated two weak points in the design. The one-inch studs were barely adequate and the energy absorbing egg crate on the base was too light. These points were modified on the Phase II design shown in figure 16. The studs on this version of the cask are 17-4 PH stainless steel, condition H1075, with a yield strength of 125,000 psi, an ultimate strength of 170,000 psi, and an elongation of 13% in two inches.²³

Throughout the development of the cask from concept to Phase II design, calculated values were checked against data obtained from the quarter scale model. In some cases no calculations were made, but rather a design change was made based on model test alone. This was the case with the modifications to the base egg crate. Comprehensive model tests indicated the most damaging position of impact was with the cask center of gravity above the closure edge. It was known that the demonstration cask would be dropped in this position and not on the bottom, but the bottom of the cask was modified to meet program objectives based on the quarter scale test information. The bottom modifications were: (1) movement of the drain valve to a more protected area in the center of the bottom plate and (2) addition of eleven crushable members to the original eight members of the bottom egg crate.

The quarter scale tests indicated that up to 300 g peaks could be experienced if the demonstration cask peaks could be scaled directly from the quarter scale data. The hypothetical acceleration time curve (figure 15) indicates peaks in the 300 g range with nothing more than superficial damage. All indications were that the cask would experience around 300 g during impact. Knowing this, and knowing a closure that is over-designed is only slightly more expensive than one that fails, a decision was reached to design the closure to remain sealed when a static force equal to 500 times the weight of the lid and cavity contents is applied to the lid in a direction that tends to remove it. For a 2500-pound lid and 3000 pounds of contents, this is equal to 1.1×10^6 lbs. Sixteen 1-1/2 inch 17-4 PH studs (stress diameter = 1.31 inches) support this load at a stress of:

$$\frac{P}{16A} = 52,000 \text{ psi} \quad (21)$$

where

$$P = 1.1 \times 10^6 \text{ lbs}$$

$$A = 0.7854 \times 1.31^2 = 1.35 \text{ in}^2$$

PHASE II DESIGN OF THE DEMONSTRATION FUEL ELEMENT SHIPPING CASK

Figure 16

DESIGN SUMMARY

A complete set of the drawings from which the demonstration cask was fabricated is included as appendix D.

The following points of interest developed during the design and fabrication of the demonstration cask. We selected 0.6 as a conservative estimate of the emissivity of weathered stainless steel. However, ordinary decontamination procedures remove the weather effect from the cask after each shipment so that the cladding should be considered as polished or sandblasted stainless steel for heat transfer calculations.

The closure should be closely examined on any cask. It should provide ease of operation under normal conditions and meet the minimum standards of leakproof operation under the hypothetical accident conditions. The closure on the demonstration cask was leakproof at the expense of ease of operation. We feel this is preferable to the opposite situation where the lid provides poor containment in an accident situation. However, a balance between the two extremes can be achieved on a stud type closure by making the lid in two pieces with the bolt flange separate from the rest of the lid. This allows easy insertion of the heavy shielded plug with no interference from the bolt flange. If it is desirable to make the lid all in one piece, a generous clearance should exist between the studs and holes. The annular clearance between the shielded section of the lid and the cask should be kept small in order to prevent circular bending of the cask and/or lateral motion of the lid relative to the cask which may damage the gasket.¹⁵

The demonstration cask design program was different from other cask design programs since it was known from the start that the cask would be subjected to normal transport and accident tests. This eliminated any tendency on the part of the designer to "shade" the regulations in favor of the cask based on an assumption that the cask "probably would never see an accident anyway." By the same reasoning, we may have considered the normal operations of loading, unloading, etc., too lightly since we knew the cask would probably never be operational. Through analysis and model testing, we were sure the cask would meet all the regulations without exception before the testing began. The probable effect of each accident event on the final design based on calculations and model tests is summarized as follows:

1. Free Drop- Most of the impact energy will be absorbed in plastic deformation of sacrificial external members when the cask is dropped in any orientation. Any energy not absorbed by the fins in the closure edge drop will be absorbed by deformation of the stud protection flange which is considered to be a part of the "solid body" center section of the cask. Expected peak acceleration of the cask in a closure drop is 300 g. Lid accelerations may be higher due to the unknown effect of the contents on the lid. Accelerations on the simulated fuel element could vary from 100 g to 300 g depending on the exact conditions at impact.

2. Puncture- Extensive calculations were not carried out on the puncture drop because the quarter scale model puncture test indicated there would be no damage resulting from this test. The orientation most damaging at impact was considered to be with the cask in the horizontal position and the center of gravity directly over the piston. This orientation generates circular bending, tangential, and longitudinal stresses in the cask shell that probably will be larger than the shearing stress tending to puncture the cask, depending on the length to diameter ratio of the cask. There is little chance of a mild steel piston puncturing the shell since the mechanical properties of the cask and piston are very nearly the same. However, as mentioned earlier, uranium has very low elongation to failure in the tensile test, and unless the load is distributed over a large portion of the cask, the tensile stress on the inside of the uranium shell could result in fracture of the shell. On the demonstration cask, the entire length of the laminated wall resists the load applied by the piston, and the cask ends are so designed that they will support the cylindrical wall, keeping it round, throughout any side drop.

3. Thermal Exposure- Since the uranium shield will not melt in a 1475°F fire and since a tungsten barrier was placed between the stainless steel and uranium at all interfaces to prevent diffusion alloying, there will be no impairment of radiation attenuation because of the fire. Thermal expansion stresses are at a minimum due to the similarity of physical properties of the uranium and 304L stainless steel.

There is an interesting advantage of a laminated shell in a thermal exposure test. The space between laminations increases when the outside temperature is higher than the inside temperature. This lowers thermal conductivity and consequently the cavity temperature lags the fire temperature more than it would in a solid shell.

4. Water Immersion- Immersion in water has no effect on the cask. A chloride solution (salt water) immersion, however, could prove harmful to the cask due to chloride corrosion and accompanying stress corrosion cracking of the stainless steel cladding.

Appendix A

URANIUM METAL INFORMATION PERTINENT TO ITS USE
FOR RADIATION SHIELDING

PHYSICAL PROPERTIES

Generally speaking, attenuation of gamma radiation is best accomplished by materials of high atomic number and density. Both uranium and lead, for example, have superior shielding properties to those of iron. Since uranium is 1.69 times more dense than lead, it follows that for equivalent attenuation of gamma radiation a thinner uranium shield would be required. A comparison of pertinent chemical and physical properties of some shielding materials, shielding thicknesses of various materials, and uranium to lead ratios for a typical shielding application are presented in this appendix in tables A-1, A-2, and A-3, respectively. As noted in table A-3, a cylindrical uranium shield would weigh about 70% to 75% of a lead shield for comparable attenuation. Additional information relative to the shape and density of shielding is presented in appendix B.

In the transport of radioactive materials, it is usually necessary to provide a method of dissipating the heat which results from the decay of radioactive sources. It is beneficial to have a shielding material with a high thermal conductivity, a high specific heat, and a high melting point. The thermal conductivity and specific heat of solid lead and uranium are nearly the same; however, above the melting point of lead, uranium has twice the thermal conductivity. Uranium also has a considerably higher melting point than lead, 2071°F versus 621°F.

THERMAL CYCLING

During the course of its life, a shielding material could be subject to thermal cycling in use. Figure A-1 shows the effect of thermal cycling range and temperatures on the growth of uranium which has not been treated to prevent the occurrence of this phenomenon. Thermal cycling below 660°F presents no uranium growth problem; however, for non-water cooled casks, the maximum uranium temperature could exceed this value. Measures adopted to prevent any possibility of this occurring in cask fabrication are discussed in detail in the body of this report.

OXIDATION

Uranium oxidizes on exposure to air in a manner similar to the rusting of mild steel. However, "canning" of uranium in stainless steel or the application of a protective coating to prevent contact between uranium and air eliminates any problem of oxidation. In the event of rupture of a "canned" uranium object, the supply of air would probably be very limited and the oxidation rate correspondingly low. Data on the effect of temperature on the corrosion rate of uranium in air are shown in table A-4. During the course of the thermal exposure test on the demonstration

Table A-1

PHYSICAL AND CHEMICAL PROPERTIES OF SOME SHIELDING MATERIALS

	Uranium	Lead	Iron
Atomic weight	238.03 ^a	207.19 ^a	55.847 ^a
Atomic number	92 ^a	82 ^a	26 ^a
Density at 20°C, g/cc	19.05 ^b	11.34 ^b	7.86 ^b
Color	Silvery metal	Silvery metal	Silvery metal
Melting point: °C	1133 ^b	327 ^b	1535 ^b
°F	2071	621	2795
Boiling point: °C	3927 ^k	1515 ^b	3000 ^b
°F	7101	2759	5432
Oxidation states	+3, +4, +6, +5 (unstable)	+2, +4	+2, +3
Thermal conductivity, cal/(cm ²)(sec)(°C)/cm			
at: 100°C	0.063 ^c	0.082 ⁱ	0.19 ^j
330°C	0.078 ^c	0.039 ⁱ	-
Specific heat, cal/(g)(°C)	0.028 ^e	0.031 ⁱ	0.108 ^h
Electrical conductivity, (ohm-cm) ⁻¹	2 to 4 x 10 ^{4f}	4.6 x 10 ⁴	17 x 10 ⁴
Hardness, Brinell scale	185-385* ^k	5 ^g	90-310* ^h
Ultimate tensile strength, psi	50-200,000* ^d	2,500 ^g	38-135,000* ^h
Minimum yield strength at 0.2% offset, psi	25,000 ^d		

* Varies with treatment.

^a Fisher Scientific/Periodic Chart of the Elements, Cat. 5-702, 1962.

^b West, R.C., et al., Handbook of Chemistry and Physics, 45th Ed., The Chemical Rubber Co. (1964).

^c Holden, A.N., Physical Metallurgy of Uranium, Reading: Addison-Wesley Publishing Co., Inc. (1958) p 46.

^d Ibid., pp 66-73.

^e Ibid., p 47.

^f Ibid., p 53.

^g Perry, R.H., et al., Chemical Engineers' Handbook, 4th Ed., New York: McGraw-Hill Book Co., Inc. (1963) pp 23-43.

^h Ibid., pp 23-31.

ⁱ Kirk, R.E., and Othmer, D.F., Encyclopedia of Chemical Technology, Vol 8, New York: The Interscience Encyclopedia, Inc., (1952) p 217.

^j Ibid., p 20.

^k Donahue, J.E., and Higinbotham, A.E., Uses of Depleted Uranium, Oak Ridge: Massachusetts Institute of Technology Engineering Practice School (MIT-OR-5).

Table A-2

BROAD BEAM GAMMA-RAY ATTENUATION^a
(With Buildup)*

Isotope	Shield Material	Inches of Shield Material					
La ¹⁴⁰	Water	26	46	62	83	106	----
	Aluminum** . .	10.5	19	28	35	43	50
	Iron	3.6	6.6	9.6	12.2	15.2	----
	Lead	1.8	3.7	5.5	7.4	9.4	----
	Uranium	1.1	2.1	3.2	4.2	5.3	6.3
Co ⁶⁰	Water	26	44	62	77	98	----
	Aluminum** . .	10.6	18	26	32	39	46
	Iron	3.6	6.2	8.8	11.1	13.7	----
	Lead	1.8	3.4	4.8	6.5	8.3	----
	Uranium	1.0	1.9	2.7	3.5	4.4	5.2
Ba ¹⁴⁰ +La ¹⁴⁰	Water	25	45	65	82	105	----
	Aluminum** . .	9.7	19	27	35	45	50
	Iron	3.4	6.4	9.2	12.0	14.9	----
	Lead	1.6	3.6	5.3	7.2	9.2	----
	Uranium	0.96	2.1	3.1	4.1	5.2	6.2
Ra ²²⁶ (In equilibrium with drs, encased in 0.5 mm Pt-Ir).	Water	25	44	64	81	104	----
	Aluminum** . .	9.8	18	27	35	44	51
	Iron	3.4	6.4	9.2	11.8	14.8	----
	Lead	1.6	3.4	5.2	7.1	9.0	----
	Uranium	0.87	2.0	3.0	4.1	5.2	6.3
Attenuation factor		10	10 ²	10 ³	10 ⁴	10 ⁵	10 ⁶

*Calculated from data of Fano (as plotted in AERE HP/I23) and recent values for spectra energies, relative intensities, and attenuation coefficients. Figures are probably conservative as buildup values were calculated for inside an infinite shield.

**Reasonable approximation for ordinary concrete.

^a Dummer, Jerome E., Jr., General Handbook for Radiation Monitoring, LA-1835 (3rd ed.), Washington: U. S. Government Printing Office (1959) p 153.

Table A-3

COMPARISON OF LEAD AND URANIUM
CYLINDRICAL CASK SHIELDS(Based on an inside radius of 5.9 inches and
a radial thickness of 11.8 inches of lead.)

Gamma Ray Energy, Mev	Uranium-To-Lead Ratios		
	Shield Thickness	Shield Volume	Shield Weight
1	0.539	0.416	0.698
2	0.564	0.442	0.741
3	0.566	0.444	0.745
4	0.568	0.446	0.748
5	0.571	0.449	0.754
6	0.571	0.449	0.754
8	0.571	0.449	0.754
10	0.571	0.449	0.754

Calculated from data presented in ANL-6000.^a

Table A-4

EFFECT OF TEMPERATURE ON CORROSION
OF URANIUM IN AIR^b

Temperature, °F	Penetration, inches/year
68	nil
212	0.10
280	0.20
392	1.98
572	3.96
752	21.38
932	28.54

^a Grotenhuis, M., Lecture Notes on Reactor Shielding, Lemont: Argonne National Laboratory (March 1959), ANL-6000.^b Wilkinson, W. D., Uranium Metallurgy, Vol 2, New York: Interscience Publishers (1962), p 802.

REFERENCE: Holden, A. N., Physical Metallurgy of Uranium, Reading: Addison-Wesley Publishing Co., Inc. (1958) p 185

THERMAL-CYCLING RANGE AND TEMPERATURES EFFECT ON GROWTH OF URANIUM

Figure A-1

cask, massive pieces of bare uranium were subjected to the same environment, i.e., 1475°F, for 30 minutes. The loss of uranium due to surface oxidation averaged less than 0.050 inch.

URANIUM AVAILABILITY

Large supplies of uranium depleted in the ^{235}U isotope are available as UF_6 for conversion to uranium metal which can then be fabricated into various shapes as may be desired. The UF_6 may be procured from the U. S. Atomic Energy Commission at a base cost of about \$1.13 per pound of contained uranium.

EFFECT OF A DEPLETED URANIUM SHIELD ON THE CRITICALITY OF IRRADIATED FUEL

A question is frequently raised concerning whether the fissile content of a depleted uranium shield would increase the criticality problems of shipping irradiated fuel elements. An evaluation indicates that the low ^{235}U content of depleted uranium metal (approximately 0.2 wt % ^{235}U) does not increase these problems. Experimental data from studies of a uranium metal lattice in water indicate that the neutron reflective properties of lead, uranium, and water to be similar.

A piece of uranium metal of unlimited size cannot support a chain reaction unless enriched in ^{235}U to a concentration of about 5 percent. Consideration of the multiplication factor of massive uranium metal in a system in which the uranium is exposed to any neutron spectrum, including all thermal neutrons, indicates that the uranium metal cannot be a multiplication medium at an enrichment less than 0.36 wt % ^{235}U .

But a more significant factor is the reflective property of massive uranium. Experimental determinations of the relative effect of lead, water, and uranium metal have been made for specialized cases. Results reported from Los Alamos²⁴ indicated that four inches of uranium is a better reflector for a uranium metal sphere or cube enriched to 93 wt % ^{235}U than a thick water or lead reflector. A similar result was observed for a highly enriched and concentrated HFIR core fuel element in which lead was found to be a somewhat better reflector than water.²⁵ However, for the more comparable case of a lattice of enriched uranium metal in water, a Y-12 report²⁶ indicated that water, lead, and normal uranium metal were comparable reflectors. The lead was slightly more reflective than water and water slightly more reflective than normal uranium.

With either lead or uranium shielding, the effect of a thick, dense reflector on the reactivity of the cask contents will have to be considered at the time the cask is designed or loaded.

URANIUM METAL HEALTH HAZARD CONTROL

Depleted uranium is radioactive but has a low specific activity. Specific alpha activity varies according to ^{235}U depletion, from 1.5 dpm per microgram for natural uranium to 0.75 dpm per microgram for pure ^{238}U . The

specific activity of beta emitting daughters is 1.5 dpm per microgram of depleted uranium. The resultant beta dose rate at the surface of uranium metal is about 240 mr per hour. Thus, uranium metal can be hand-held for 40 hours without exceeding the quarterly Radiation Protection Guide of 10 rem to the skin. Gamma radiation is emitted coincident with a fraction of both alpha and beta radiation and results in an exposure dose rate of about 6 mr per hour at the surface of massive uranium metal pieces. Radiation readings taken five feet above the surface of 200 - 300 tons of uranium derbies covering an area roughly 20 feet by 14 feet were 5 mr per hour gamma and 85 mr per hour beta plus gamma. The beta contribution can be shielded by about 1/8 inch of steel.

Uranium metal exposed to the atmosphere oxidizes readily. The bright silver-white machined surface almost immediately takes on a bronze hue. This surface can be preserved for several months by spraying with a light coat of plastic varnish. Unless the surface is protected from the atmosphere, it darkens and turns black in a few days. Some of the black oxide can then be removed by rubbing with slight pressure, and handling the metal results in loose oxide contamination. Uranium metal descaled by dry honing (shot blasted) and plastic sprayed turns black quickly, but rubbing does not readily remove the oxide.

The uranium oxide presents a minor problem for control because of the possibility of its becoming airborne and being inhaled. The current MPC air values of the NCRP as published in NBS Handbook 69 are equivalent to 0.18 mg/m³ of air. This may be compared to the 0.25 mg/m³ threshold limit value for insoluble compounds of uranium and the 0.2 mg/m³ threshold limit value for lead published by the American Conference of Governmental Industrial Hygienists.

...the ... of ...
...the ... of ...
...the ... of ...
...the ... of ...
...the ... of ...

...the ... of ...
...the ... of ...
...the ... of ...
...the ... of ...
...the ... of ...
...the ... of ...
...the ... of ...
...the ... of ...

...the ... of ...
...the ... of ...
...the ... of ...
...the ... of ...
...the ... of ...
...the ... of ...
...the ... of ...
...the ... of ...

Appendix B

FACTORS AFFECTING THE WEIGHT RATIO OF URANIUM TO LEAD
FOR RADIATION SHIELDING

If the generalization is used that radiation attenuation is strictly a function of mass, i.e., for equal frontal areas, a pound of any material will provide the same amount of shielding as a pound of any other material; then the relative mass or thickness for lead and uranium shields of various shapes can be determined. If, for example, the shield is a flat wall of specific area, then the mass ratio between uranium and lead which will provide equal shielding is unity. The thickness of a uranium wall having the same mass as a lead wall would be less than that of a lead wall since uranium is more dense than lead. The actual thickness ratio would be the reciprocal of the ratio of the respective material densities. If the density of the lead is assumed to be 11.3 and the density of the uranium is assumed to be 19.0, then the ratio of equivalent wall thickness would be $11.3/19.0$ or 0.595. This indicates that a uranium wall 0.595-inch thick would be equal in shielding to a lead wall 1.0-inch thick, and the two walls would be equal in weight.

If the shield being considered is in the shape of a sphere with a very small cavity, then the mass ratio between equivalent uranium and lead shields is 0.35. This ratio can be determined by calculating the weights of uranium and lead spheres having their radii in proportion to 0.595. The cases of the flat wall and the sphere with an extremely small cavity represent the extreme shape factor conditions. In the more usual case where the shield is in the shape of a hollow cylinder with end plugs, mass ratios in the order of 0.75 are typical. Some calculations which illustrate the shape-factor effect of different shaped shields are given in table B-1. These calculations do not consider the relative shielding efficiencies of the two materials. They assume a constant thickness ratio for all shapes and are presented only to illustrate how cask geometry affects the mass ratio. Factors used to make these calculations include:

1. The thickness ratio to provide equal shielding is 0.595, i.e., the density of lead (11.3) divided by the density of uranium (19.0).
2. Uranium weighs 1187 lbs/ft³.
3. Lead weighs 706 lbs/ft³.
4. All measurements quoted are in feet.

Since the shielding ability of uranium, on an equal mass basis, is somewhat better than lead, the actual thickness ratio of uranium to lead to obtain equal radiation attenuation is somewhat smaller than is indicated by their density ratio.

The actual mass ratio also varies with the specific energy of the radiation being absorbed. Table B-1 lists the actual mass ratios of equivalent uranium and lead shields of a specific size assuming various source energy levels.

Table B-1

EFFECT OF SHAPE FACTOR ON
URANIUM-LEAD WEIGHT RATIO FOR SHIELDING

Case	Shield Shape	Size, Feet	Weight, Pounds	Ratio, Wt U/Wt Pb
1	U Wall	3.0 x 3.0 x 0.595	6,362	1.00
	Pb Wall	3.0 x 3.0 x 1.0	6,354	
2	U Hollow Cylinder	Exterior: 3.19 dia x 11.19 long Interior: 2.0 dia x 10.0 long	68,762	0.82
	Pb Hollow Cylinder	Exterior: 4.0 dia x 12.0 long Interior: 2.0 dia x 10.0 long	84,240	
3	U Hollow Cylinder	Exterior: 2.19 dia x 11.19 long Interior: 1.0 dia x 10.0 long	40,690	0.75
	Pb Hollow Cylinder	Exterior: 3.0 dia x 12.0 long Interior: 1.0 dia x 10.0 long	54,310	
4	U Hollow Sphere	Exterior: 2.19 dia Interior: 1.0 dia	5,923	0.62
	Pb Hollow Sphere	Exterior: 3.0 dia Interior: 1.0 dia	9,623	

Appendix C*

ALLOYING OF URANIUM AND STAINLESS STEEL
IN URANIUM SHIELDED CASKS

ABSTRACT

Solid state diffusion will occur between uranium and stainless steel when in contact at temperatures as low as 1000°F. It is shown here that this diffusion rate is significantly temperature dependent and that at 1475°F is sufficiently rapid to form a low melting alloy that will flow away from the contact area within one hour. Combination of the two metals has been prevented by the use of diffusion barriers between the stainless steel and the uranium. Copper is a useful barrier up to 1750°F and tungsten is effective up to at least 1850°F.

INTRODUCTION

Depleted uranium metal is being proposed as a shielding material in the fabrication of casks for shipping radioisotopes and for containers for radioactive heat sources. Since uranium metal has low resistance to corrosion, it must be protected by a coating or be canned in a corrosion-resistant material. Stainless steels have been chosen for this use because of their resistance to oxidation at high temperatures and their resistance to attack by most of the common cleaning agents. Previous investigators have shown, however, that uranium combines with stainless steel by solid state diffusion at temperatures above 1000°F. The iron-uranium eutectic melts at 1337°F so that if these two materials are in intimate contact at this temperature a molten alloy will be formed.¹

The possibility exists that shipping casks could be subjected to temperatures in excess of 1337°F if accidentally involved in a fire. Therefore, tests were conducted to determine at what temperature the alloying would result in catastrophic loss of the stainless steel cladding.

PAST WORK

Previous investigators have shown that uranium in contact with stainless steel will penetrate the stainless steel by solid state diffusion in 24 hours at 1400°F (760°C). At 1355°F (735°C) there was no attack on the stainless steel. A tungsten crucible was not attacked by molten uranium at 2372°F (1300°C) in 4 hours.²

CURRENT PROGRESS

The objectives of the current work were to define the magnitude of alloying between uranium and stainless steel as it pertained to uranium shielded

*The information presented here is taken in its entirety from Paducah Plant Laboratory Progress Report, KY-L-419, Part 1, issued September 27, 1966.

casks and to provide an adequate diffusion barrier between the uranium and stainless steel for uranium shielded casks.

Test specimens were made up as shown in figure 1 to represent the condition that would exist with stainless steel cladding on a uranium cask. In this case it represented a cask fabricated from 1/2-in. thick laminates of depleted uranium plate. The cladding materials were 1/4-in. thick type 405 stainless steel for the outer clad and 3/16-in. thick type 304 stainless steel for the inside liner.

In the fabrication of casks, the air would be evacuated from between the laminates and the void volumes backfilled with helium. The first tests were made to determine if a significant difference exists when the samples were heated in a helium atmosphere or in a vacuum. The results indicate no significant differences; thus, the following specimens were tested by heating for one hour in a vacuum furnace with a pressure of less than 1.0 micron.

Sample 1 - No diffusion barrier between uranium and stainless steel.

Sample 2 - Stainless steel surfaces electroplated with copper 0.005-in. thick.

Sample 3 - Copper sheet (0.010-in thick) between the stainless steel and the uranium.

Sample 4 - Stainless steel surfaces plasma flame spray coated with 0.005-in. thick coating of tungsten.

All of the uranium surfaces were dry honed as were the stainless steel surfaces where no barrier coating was used. The surfaces of the coated stainless steels were tested as coated.

The sample which contained no diffusion barrier showed that both stainless steel types 304 and 405 would combine with uranium by solid state diffusion. When this sample was heated at 1475°F for one hour, a low melting alloy formed and flowed away from the contact areas (figure 2). The amount of metal that flowed increased as the test temperature was raised to 1750°F. The formation of the lower melting alloy did not extend beyond the first 1/2-in. thickness of uranium. This is believed to be due to the inability of the stainless steel components to diffuse beyond the superficial film on the surfaces of the uranium between the laminates. The alloying was such that the components of the steel diffused into the uranium, and although the first laminate of uranium was penetrated the stainless steels were not. However, these cladding materials were attacked severely where the low melting alloy came in contact with them at the bottom of the sample.

Either copper sheet or copper electroplate on the stainless steel prevented catastrophic alloying when these samples were heated at 1750°F for one hour (figure 3). However, when these samples were heated at 1850°F for one hour, the amount of alloying was the same as that which occurred in the unprotected samples (figure 4).

LEGEND

TYPE 405 STAINLESS STEEL

DEPLETED URANIUM

TYPE 304 STAINLESS STEEL

ALLOYING TEST FIXTURE

Figure C-1

Figure C-2

660223-2(U)

No diffusion barrier. Heated one hour at 1475°F.

Figure C-3

660221-2(U)

Stainless steel surfaces electroplated with 0.005-in. copper. Heated one hour at 1750°F.

Figure C-4

660224-9(U)

Stainless steel surfaces electroplated with
0.005-in. copper. Heated one hour at 1850°F.

Figure C-5

660224-6(U)

Stainless steel surfaces plasma flame coated with
0.005-in. tungsten. Heated one hour at 1850°F.

The samples containing plasma flame sprayed coatings of tungsten on the surfaces of the stainless steels in contact with the uranium showed that this coating is an effective barrier to the diffusion at 1850°F (figure 5).

These tests have shown that at temperatures much above the alloy melting point, 1337°F, extensive damage would result to a cask from the formation of a uranium-iron alloy. Both the structural integrity and shielding efficiency would be affected. A copper diffusion barrier will provide some protection, being effective up to 1750°F. At 1850°F a plasma sprayed tungsten barrier effectively prevented diffusion and alloy formation. Tests at temperatures above 1850°F were not attempted, but it is believed that the tungsten barrier would provide protection at temperatures near the melting point of uranium (2070°F). There was little difference in the alloying reaction between the 304 and 405 stainless steels.

REFERENCES

- ¹ Kaufman, Albert R., Nuclear Reactor Fuel Elements: Metallurgy and Fabrication, New York: Interscience (1962), p 248.
- ² Tipton, C. R. (ed), Reactor Handbook, Vol I, 2nd ed., New York: Interscience (1960), pp 478-480.

Figure D-1

Figure D-2

Figure D-3

Figure D-4

Figure D-5

Figure D-6

Figure D-7

Figure D-8

Figure D-9

REFERENCES

- ¹ Schedule 189a, Justification for Operating Costs and Equipment Obligations for Reactor Development Programs, No. 24 0019 001 (June 16, 1966).
- ² Smith, V.A., and Loveland, C.W., Demonstration Fuel Element Shipping Cask from Laminated Uranium Metal. Part II. Fabrication and Cost, UCCND, Paducah Gaseous Diffusion Plant (March 20, 1969) (KY-551).
- ³ Clifford, C.B., Demonstration Fuel Element Shipping Cask from Laminated Uranium Metal. Part III. Tests and Demonstrations, UCCND, Paducah Gaseous Diffusion Plant (March 12, 1969) (KY-552).
- ⁴ Shappert, L.B., and Salmon, Royes, Shipping Cost Comparisons for Lead, Steel, and Uranium Shielded Casks, UCCND, Oak Ridge National Laboratory (March 1966) (ORNL-3918).
- ⁵ Thompson, J.T., and Morgan, J.M., "Nuclear Transportation," Nuclear Safety, Vol. 8, No. 5 (1967), pp. 443-454.
- ⁶ Clifford, C.B., Design and Fabrication of a Prototype Laminated Uranium Metal Shipping Cask for Large Shipments of ^{60}Co , UCCND, Paducah Gaseous Diffusion Plant (April 3, 1967) (KY-521).
- ⁷ Shewmon, Paul G., Diffusion in Solids, New York: McGraw-Hill Book Company (1963).
- ⁸ Clifford, C.B., The Design, Fabrication and Testing of a Quarter Scale Model of the Demonstration Uranium Fuel Element Shipping Cask, UCCND, Paducah Gaseous Diffusion Plant (June 10, 1968) (KY-546).
- ⁹ Tipton, C.R., Jr., (ed.), Reactor Handbook (2nd ed.), Vol 1, New York: Interscience Publishers, Inc. (1960), pp. 90-146.
- ¹⁰ Harrington, C.D., and Ruehle, A.E., Uranium Production Technology, New Jersey: D. Van Nostrand Company, Inc. (1959), pp. 294-366.
- ¹¹ Holden, A.N., Physical Metallurgy of Uranium, Reading: Addison-Wesley Publishing Company (1958), pp. 35-58 and 181-195.
- ¹² Rossmassler, W.R. (ed.), Paducah Plant Laboratory Progress Report, UCCND, Paducah Gaseous Diffusion Plant (December 1968) (KY-L-469, Part 1).
- ¹³ "Title 10, Code of Federal Regulations, Part 71," Federal Register, 31, No. 141 (Friday, July 22, 1966).
- ¹⁴ Personal communication between R. W. Peterson of National Lead Company and L. B. Shappert of Oak Ridge National Laboratory.
- ¹⁵ Shappert, L.B., A Guide to the Design of Shipping Casks for the Transportation of Radioactive Material, UCCND, Oak Ridge National Laboratory (April 1965) (ORNL-TM-681).

- ¹⁶Jakob, M., and Hawkins, G.A., Elements of Heat Transfer, (3rd ed.), New York: John Wiley and Sons (1957).
- ¹⁷Metals Handbook, Vol. 1, Ohio: American Society of Metals (1961).
- ¹⁸Keenan, J.H., and Keyes, Frederick, Thermodynamic Properties of Steam, New York: John Wiley and Sons (1936).
- ¹⁹Parker "O" Ring Handbook, Catalog OR-5700, Parker Seal Company (April 1967).
- ²⁰Higden, A., and Stiles, W.B., Engineering Mechanics, Vol. 1 (2nd ed) New Jersey: Prentice-Hall (1957).
- ²¹Seely, F.B., and Smith, J.O., Advanced Mechanics of Materials (2nd ed.), New York: John Wiley and Sons (1932).
- ²²Nelson, V.G., The Mechanical Properties of Wrought Depleted Uranium and of Welds Joining This Material, Sandia Corporation (March 1961) (SC-DR-58-61).
- ²³Armco Precipitation Hardening Stainless Steels, Armco 17-4 PH Bar and Wire, Product Data Bulletin, Ohio: Armco Steel Corporation (November 3, 1958).
- ²⁴Paxton, H.C., Thomas, J.T., Callihan, A.D., and Johnson, E.B., Critical Dimensions of Systems Containing ^{235}U , ^{238}Pu , and ^{233}U , Los Alamos Scientific Laboratory and Oak Ridge National Laboratory (June 1964) (TID-7028).
- ²⁵Callihan, A.D., Cronin, D.F., Fox, J.K., Morfitt, J.W., Rohrer, E.R., and Williams, D.V.P., Critical Mass Studies, Part VI, UCCND, Oak Ridge National Laboratory (August 1951) (Y-801).
- ²⁶Neutron Physics Division Annual Progress Report, Vol. 1, UCCND, Oak Ridge National Laboratory (November 1965) (ORNL-3858), p. 18.

ACKNOWLEDGEMENTS

The authors acknowledge the assistance and cooperation received from others in the areas of technological development, engineering design, and overall project coordination relative to the demonstration fuel element shipping cask. The following Paducah Plant personnel made valuable contributions to the program: R. C. Baker, C. B. Clifford, W. R. Golliher, E. Gottschalk, R. Johnston, R. W. Levin, W. R. Pedigo, B. D. Penry, W. R. Rossmassler, and V. A. Smith.

Gratitude is also extended to W. P. Gammill and J. A. Sisler of the U. S. Atomic Energy Commission and L. B. Shappert of the Oak Ridge National Laboratory.