Proceedings of a Workshop on Uses of Depleted Uranium in Storage, Transportation and Repository Facilities

Las Vegas, Nevada
July 15-17, 1997

U.S. Department of Energy
Assistant Secretary for Environmental Management
Office of the Deputy Assistant Secretary for Science and Technology
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Washington, DC 20585
Department of Energy participants from the Office of Civilian Radioactive Waste Management, Yucca Mountain Site Characterization Project Office, Office of Nuclear Energy, and Office of Environmental Management cooperatively sponsored a workshop on possible uses of depleted uranium related to storage and disposal of spent nuclear fuel and high-level radioactive waste. This record of the proceedings of the workshop is authorized for publication as a resource document for planning future programs.

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FOREWORD

A workshop was held July 15-17, 1997 in Las Vegas to discuss possible uses of depleted uranium (DU) products from currently stored uranium hexafluoride (UF6). Uses were reviewed that specifically supported storage, transport, and disposal of DOE high-level waste and civilian spent nuclear fuel. A number of promising options were identified and acknowledged involving radioactive shielding or chemical enhancements to delay migration; however, the discussion of associated costs was generally considered to be beyond the scope of the workshop and was left for subsequent studies.

The majority of the Department’s depleted uranium is currently in the form of depleted uranium hexafluoride (UF6), approximately 560,000 metric tons. All of the options considered at the workshop documented in these Proceedings will require conversion from UF6 into a form suitable for potential use in a repository. Present estimates of the potential cost to convert and dispose of conversion products is $1.6-$3.9 billion. With this magnitude of costs, finding beneficial uses of DU products rather than disposal is justified.

Significant mutual benefits may accrue to the Office of Civilian Radioactive Waste Management (OCRWM), the Depleted UF6 Program, and the Environmental Management Program if depleted uranium products were used in the repository rather than procuring commercially available materials that provide similar repository performance enhancements. Workshop participants agreed that the cost advantage or disadvantage must now be established. Likewise, costs to the user, e.g., OCRWM, for a particular use of DU products was not yet considered. Needed cost tradeoffs for OCRWM to use depleted uranium, as opposed to a commercially-available material, were not fully addressed at the workshop. Therefore, any final decision on the use of DU products by OCRWM and the Department must consider further assessments. Subsequent assessments should describe how the Department will pay for the conversion and fabrication of DU into a useable product and the economic tradeoffs to the user and others for particular applications compared to similar materials that can perform the same function as DU products.

The overall system cost for DU product use involves the accumulation of cost tradeoffs involving the civilian nuclear power industry, DOE Environmental Management, DOE Nuclear Energy, and, OCRWM if DU products were used by OCRWM in dry storage of spent fuel and/or in the repository. Such tradeoffs relate to the various cost elements for the use of DU products and the costs incurred by the various organizations. These are yet to be done on a comparable basis.

In December, 1997, the Office of Nuclear Energy, Science and Technology issued the “Draft Programmatic Environmental Impact Statement for Alternative Strategies for the Long Term Management and Use of Depleted Uranium Hexafluoride”, DOE/EIS-0269. The preferred option is to use DU products rather than disposal of products from UF6 conversion. A separate cost report was issued in support of the PEIS by Lawrence Livermore National Laboratory, UCRL-AR-127650. Since then, industrial sources have suggested options, yet unproven, for significant cost reductions in the conversion process using alternative technologies.
EXECUTIVE SUMMARY

Representatives from three DOE offices and their support contractors attended a workshop July 15-17, 1997 in Las Vegas to discuss the possible use of depleted uranium (DU) in the DOE high-level waste and spent nuclear fuel repository program. Presentations were made on the plans and progress of the repository program, the management of the large DU resource owned by the Department, possible uses of DU in the repository program, and the progress of one alternative for reuse and recycling the DU resource as shielding.

Workshop attendees concluded that while DU is not required for the success of the repository program, beneficial uses should be evaluated. Given the overall needs of the Department, and the tremendous potential cost of doing anything with the DU (except for maintaining its present stored status), DOE decision makers should be aware of alternatives for beneficial uses -- and the work required to determine their costs and benefits.

After the first day, the workshop broke into two work groups -- “pre-closure” and “post-closure” groups -- to discuss the possible advantages and disadvantages of eighteen potential repository DU use options during the repository period associated with their time period.

The pre-closure group recommended that options involving DU in metal or DUCRETE form (concrete made with DU aggregate) in shielded casks and in horizontal half-cylindrical shields that cover the waste package after emplacement warranted further study. Theoretical analysis have shown DUCRETE concrete to have better attenuation properties with a reduced weight compared to conventional concrete container. Conceptually, the DU metal or DUCRETE cask could be used throughout the fuel cycle – for at reactor storage, for interim storage at the future centralized storage facility, and as a shield for the waste package in the repository.

The post-closure group recommended further work be done to answer questions and examine performance on options utilizing DU inverts beneath the waste packages in the repository drifts (material to make a flat surface for each waste package) and DU as a potential backfill around the waste package.

If DOE decides that DU materials are to be seriously considered in spent fuel storage applications such as storage casks at reactors and interim storage facilities, detailed designs should be initiated immediately and casks tested, or DU will be precluded as an option by missing the time of peak demand for such casks. Transportation and in-repository possible applications are also approaching decision points wherein lack of performance and characteristic data will preclude further consideration of this resource if steps are not taken soon to fully evaluate promising alternative uses.

The workshop enhanced communications among the three DOE offices involved, caused the production of several relevant white papers on various topics, and attendees urged that such interactions continue.
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1. INTRODUCTION

A workshop on the potential uses of depleted uranium (DU) in the repository was organized through the cooperative efforts of the Office of Civilian Radioactive Waste Management (OCRWM), the Office of Nuclear Energy Science and Technology (NE), the Yucca Mountain Site Characterization Office (YMSCO), and the Office of Environmental Management (EM) to coordinate the planning of future activities. The attendees, the original workshop objective and the agenda are provided in Appendices A, B, and C. After some opening remarks and discussions, the objectives of the workshop were revised to:

1. exchange information and views on the status of the Department of Energy (DOE) activities related to repository design and planning;
2. exchange information on DU management and planning;
3. identify potential uses of DU in the storage, transportation, and disposal of high-level waste and spent fuel; and
4. define the future activities that would be needed if potential uses were to be further evaluated and developed.

This summary of the workshop is intended to be an integrated resource for planning of any future work related to DU use in the repository. The synopsis of the first day’s presentations is provided in Appendix D. Copies of slides from each presenter are presented in Appendix E.

The workshop did provide an opportunity to consider the possible integration of activities between DOE programs that will involve the future expenditure of billions of dollars before the use and/or disposition of DU will be accomplished. Each of the offices gained a better understanding of the technical, economic, and regulatory status of the each others’ programs, as well as an understanding of the scope of support that would be needed to make the effort successful.

This report is not intended to be a commitment that work on DU uses will be conducted by any of the participating DOE organizations.

2. BACKGROUND

Starting in 1992, after exploring a number of alternatives (Reference 1), EM focused on development and evaluation of concepts for use of DU as radiation shielding (References 2 and 3). Use of DU as shielding was considered attractive because it offers the potential of using all of the DU currently stored in various forms by the Department. Approximately 560,000 metric tons of DU exist as...
uranium hexafluoride (UF₆). In addition, 10,000 MT is uranium metal and 22,000 MT is in oxide form (Reference 4). Both metallic and oxide forms of DU could be used in shielding applications. The EM program focused on the development of DUCRETE concrete because of the high cost of fabricating shielding with DU metal. The Idaho National Engineering and Environmental Laboratory (INEEL) developed DUCRETE¹ concrete as a concept in which the oxide forms of DU were converted to a ceramic aggregate (DUAGG)¹ which, when combined with cement, formed a very dense concrete for effective gamma and neutron shielding. Experimental work on DUCRETE continued through FY 1997 with the production of test quantities of DUAGG through an industrially co-funded contract.

For the last four years, NE has been evaluating the long-term planning options for DU management, including preparing a Programmatic Environmental Impact Statement (PEIS) on the DU stored as solid UF₆ in cylinders located at the gaseous diffusion enrichment plants in Oak Ridge, Tennessee; Paducah, Kentucky; and Portsmouth, Ohio. This DU was produced by the NE uranium enrichment program for civilian and defense purposes. The PEIS will cover the options for managing the DU including options for continued storage, re-use, and disposal. As part of the evaluations, NE solicited ideas from the public through a request in the Commerce business daily. The ideas were received and evaluated preparatory to initiating the EIS (Reference 5).

For very long-term storage or disposal options, the UF₆ must be converted to an oxide. From the conversion process uranium oxide and fluorine compounds are produced. Ideally, the fluorine byproduct, hydrogen fluoride, will help offset the cost of conversion if kept relatively free of uranium. NE and EM are co-funding with industry a project to design, build, and operate a pilot scale demonstration plant to convert over 130,000 pounds of UF₆ to uranium oxide products by July 1998.

For beneficial reuse, options have been considered which use the metallic and oxide forms of uranium. Key to future large scale deployment will be the costs for converting the material into its final form, metal or oxide. To further investigate and develop applications, NE initiated in FY 1997 a project entitled “Beneficial Uses of Depleted Uranium.” The philosophy behind the project is to promote the economical use of DU.

Following resolution of technical and cost issues associated with DU use, policy and regulatory issues associated with commercial use of the DU product must also be resolved. Specific technologies and re-use options being developed under this NE project are being coordinated with previous EM development activities (DUCRETE and DUAGG).

For several years, DOE’s Office of Civilian Radioactive Waste Management (RW) has also been considering the possible beneficial use of DU in internal evaluations. This included workshops and evaluations of DU use within the mined geologic repository.

¹ DUCRETE is the trademark name applied to concrete made by replacing the conventional aggregate in concrete with a heavy aggregate made from depleted uranium oxide. DUAGG is the trademark name applied to the depleted uranium aggregate used to make DUCRETE concrete. DUCRETE and DUAGG are trade names of the Lockheed Martin Idaho Technologies Company.
3. SUMMARY

Workshop participants recognized that use of DU in the repository is only an option and is not needed to carry out the mission of the repository. The workshop concluded that although the current repository requirements can be met without the use of DU, the presence of depleted uranium products may provide advantages to repository design and operations; however, the presence of large quantities of DU may also complicate the licensing required for the repository. The costs/benefits from DU use in the repository have not been fully evaluated with respect to the total systems life cycle cost starting with present storage of DU through to final use.

3.1 The Options

The workshop focused on assessing the group of eighteen potential applications for DU conceived to enhance performance of the pre-closure and post-closure repository system operation and/or performance. These eighteen options are described in Appendix F. The DU options included variations of (1) DU use in various construction features of the repository, (2) DU use in shielding for the fuel canisters and waste packages in surface and subsurface applications, and (3) DU use for isotopic dilution to provide long-term improved repository performance (defense-in-depth) by reducing the likelihood of post-closure criticality and possible retardation of post-closure radionuclide migration.

The workshop participants concluded that, although the current repository requirements for a geologic repository can be met using the current design without DU, some of the eighteen options could provide some attractive benefits. Three applications were selected as representing the views of the participants for further evaluations:

1. **Shielded Casks**: Uranium metal or DUCRETE concrete could be used in cask designs to shield the waste packages. Such shielding would allow personnel access to the drifts for maintenance and inspection, as well as having some potential future benefit for chemical conditioning of groundwater and isotopic dilution for defense-in-depth protection against possible criticality. Further, with appropriate design of casks, waste packages would be protected from possible dripping water in the drifts. Although, emplacement of the heavier shielded waste package increased repository complexity, subsequent operation over the 100-year pre-closure period might be simplified. Repository ventilation requirements during the pre-closure period were also increased by this option.

2. **Half-Cylindrical Shield with End Ventilation**: The shields over the top of each horizontally-emplaced waste package could provide some radiation shielding benefits as well as water-drip shield benefit while allowing cooling by ventilation air. Some personnel access, chemical conditioning, and isotopic dilution benefits may also be possible.
3. **Emplacement of DUCRETE Inverts Under Waste Packages:** The material used to level the floor of the waste emplacement drifts (inverts) could be fabricated or prefabricated from DUCRETE concrete using uranium oxide aggregate (DUAGG). This option provides no shielding benefits, but does use substantial quantities of depleted uranium in the repository in a chemical form similar to that of the spent nuclear fuel. In this geometry, isotopic dilution and chemical conditioning advantages may be provided.

### 3.2 The Uncertainties

Although the possible benefits appear attractive, a number of uncertainties were recognized that require detailed evaluation. These included:

1. **Shielding (1 & 2 above):** Limitations on the peak cladding temperature would require additional drift forced ventilation or a substantial reduction in the amount of fuel in each waste package. Additional cost will be incurred for design, development, and testing required to determine if shielded waste packages are a viable option. Impacts on repository cost and schedules must also be considered. Production costs of DU metal shielded and DUCRETE casks are also uncertain.

2. **Inverts:** The management of potential contamination from the uranium could impose some operating constraints even if the DUCRETE concrete slabs are pre-fabricated and then emplaced in the drifts. The potential chemical benefits need to be verified for retardation of radionuclide migration, and for isotopic dilution effects. The durability of DUCRETE concrete in the repository chemical and thermal environment needs to be established.

The workshop recommended additional scientific, engineering, and economic studies be conducted to improve the understanding of the benefits and associated costs before any decisions could be made on the viability of the suggested concepts. Special consideration should be given to providing a basis for making programmatic decisions that would justify affecting the repository design, considering the benefits to the management of government-owned DU.

### 3.3 The Priority Needs

The high priority needs (not necessarily in order of priority) are listed below.

1. Evaluation of the heat transfer impacts of using uranium metal or DUCRETE concrete casks for radiation shielding of waste packages. Various options for inclusion as part of the waste package or as a separate external shield are possible.

2. Evaluation of impacts of various DU-shielded concepts on the drift design including mechanical emplacement options and space requirements.
3. Evaluation of the impacts of DU on overall repository performance; effects on near-field performance resulting from changes in soil and moisture chemistry that could beneficially affect radionuclide migration through the host rock; and effects on potential criticality resulting from chemical interactions between ambient rock minerals, depleted uranium, and dissolved waste constituents in near-field or far-field conditions.

4. Further confirmation of the viability of DUCRETE and DU metal casks whether used in surface storage or in the repository (costs, life-time integrity, performance).

5. Further evaluation of larger system (DU management program together with waste management program) life cycle costs under various assumptions that show cost avoidance or cost reduction opportunities balanced against possible repository cost increases.

4. WORKGROUP OBSERVATIONS SUMMARY

The following sections present a summary of the discussions held in the pre-closure and post-closure breakout sessions. The detailed discussions from both breakout sessions are documented in Appendices G and H.

4.1 Pre-Closure Workgroup Summary

1. A number of shielding concepts are possible for waste packages emplaced in the repository either for captive, self-shielded casks, or as separate additions that can serve the functions of a water drip shield as well as a radiation shield; however, no compelling single reason was agreed upon to justify using DU as a shield. The combination of using DU in shielded surface storage functions, and in later waste-emplacement functions, may have some total system life cycle cost advantages that could justify use of DU (e.g., reduction of DUF₆ management costs, lower spent fuel interim storage costs, and repository design and operation simplification).

Better integrated system costs are needed that would include costs of possible surface use of DU-shielded casks, and the subsequent potential subsurface use of the shielded casks for emplaced waste. Studies to support a DOE policy are needed that would consider the consequences to the congressionally authorized D&D and Nuclear Waste Funds from the balancing of the potential value of DU in the above applications, the potential costs of consequent changes in repository design, and any offsetting potential DU disposal costs.

2. Design constraints (e.g., allowable size and weight of casks) need to be evaluated relative to the potential use of DU to determine if present repository design is capable of incorporating DU-bearing materials or casks in the repository. Evaluation is needed of cask design requirements
for both surface and underground use. The feasibility of eventual repository disposal of used
DU containing casks and the associated incremental costs need to be understood well enough to
make a policy decision on where casks should be disposed if they were only used for storage at
surface locations.

3. Current and future regulatory effects on the use of DU are not clear and could affect choices for
management of DU (due, in part, to the substantial costs of doing anything with the large
amounts of DU existing in the DOE system). Regulations that may affect any future
disposition of DU could impact whether beneficial uses of DU should be considered. It is NE’s
position that DU should be used in the repository if the benefits justify the costs. Direct
disposal of DU in the repository is not under consideration. Workshop opinions indicated that
there is no regulation that would preclude inclusion of DU in the repository for useful purposes.
However, if DU were placed in the repository strictly for disposal purposes, then legislative and
licensing modifications would be necessary to include low-level waste. Disposition of DU in
large quantities as addressed in an NRC technical position paper\(^2\) will likely require a separate
mined facility if DU is not used in the spent fuel and high-level waste repository (included as
Appendix I). The costs of this DU disposal option could easily drive incentives to include the
shielded options in the high-level waste repository as the least expensive system option. These
opinions need to be supported by future total life cycle cost analyses of separate, mined
low-level waste disposal facilities, compared to incremental costs of inclusion in the repository
applications. In both cases, use of DU as shielding in surface casks versus DU disposal needs
to be compared to determine economic incentives.

4. Use of shielded waste packages in the repository could eliminate the need for fully-remote
handling equipment under conditions of high temperature and high radiation. However,
maintaining the current design feature of being able to remove any one waste package from a
repository drift while the others are kept in place will require accessibility evaluations. It was
further noted that waste package removal feature was not a requirement and, in fact, might not
ever be used because of associated handling risks. Cost comparison studies of shielded and
unshielded concepts would provide the needed perspective.

4.2 Post-Closure Workgroup Summary

1. DU is not required to meet post-closure performance requirements; however, DU may add to
defense-in-depth protection in two areas—external criticality and chemical buffering. The
potential for external criticality is reduced by adding DU to dilute U-235 enrichment under
postulated groundwater intrusion scenarios. If water enters the repository, chemical dissolution
of the depleted uranium by the groundwater reduces the spent fuel dissolution rate and reduces
the fissile concentration of the uranium saturated liquids exiting the waste package and

\(^2\) Letter dated 9/22/92 to Louisiana Energy Services, L.P., Attn., W. Howard Arnold (from Jerry J. Swift
for John W. N. Hickey, Chief, Fuel Cycle Safety Branch, Division of Industrial and Medical Safety, Office of
Nuclear Material Safety and Safeguards, U.S. Nuclear Regulation Commission)(included as Appendix I).
diffusing through the invert. The latter advantage appears to be more important than the former. The DU internal canister backfill option does not appear to offer clear benefits to internal criticality protection since alternative materials can do the job more economically and without impacting heat dissipation from the waste package.

2. Depleted uranium in oxide form with approximately the same dissolution and mobilization characteristics as the U-235 in spent nuclear fuel would add defense-in-depth for external criticality control. This would be beneficial, but not required, for high and medium enriched DOE-owned spent nuclear fuel. Mixing the DU with the invert and/or placing it in and around the canisters in the disposal containers would be more beneficial (from the standpoint of chemical buffering) than mixing it in backfill material. DU used as a backfill offers additional advantages of proximity and collapse resistance but with attendant drawbacks related to insertion, potentially enhanced corrosion, and potential mechanical interactions.

3. Based on the benefits and consideration of the relative effectiveness of the options in achieving the desired benefits, the use of DU in the inverts supporting the waste packages appears to offer the greatest promise. The backfill and container filler options would also place DU in relatively close proximity to the SNF, and it may be premature to eliminate these options from the preferred path. The DU would be added as an aggregate during fabrication of the inverts, thus simplifying the industrial hygiene and radiological controls issues associated with handling DU. Encapsulation of the DU in the inverts would help to bring the release of the DU to the repository environment into better congruence with the release of fissile uranium from the SNF. It would also be in the direction of gravity flow of water, and it would not negatively impact the thermal performance of the waste packages or the repository.

4. The chemistry and release of the DU in the inverts need to be investigated to ensure that DU would behave in the repository field in a manner similar to that of the uranium released from the spent nuclear fuel. Matters such as dissolution kinetics, chemical form of the DU at various stages following release, and the effect on diffusion through the rock need to be investigated further. A key goal of the investigation should be to demonstrate that the DU would be present at the time and location necessary to reduce the probability of criticality and in the proper chemical form. The thermal conductivity and heat capacity of the appropriate DU compound would have to be developed if the backfill or disposal container filler options are to be pursued. The effects of any application of DU should be incorporated into the repository Total System Performance Assessment (TSPA) to determine the overall system performance benefits or detriments prior to a definitive decision.

5. FUTURE TECHNICAL AND STUDY NEEDS

If DU materials are to be used in surface spent fuel storage applications such as storage casks at reactors and interim storage facilities, detailed designs should be initiated immediately or DU will be precluded as an option. DOE RW is actively soliciting private sector investments to provide spent fuel transportation systems to support repository operations. These transportation systems must consider
the design impact of any repository DU application. For repository only DU applications, technical information and engineering assessment information are needed before 2006, based on current likely schedules for the repository. Otherwise, many of the options are eliminated because of “locked-in” concepts and investments in engineering and designs for the repository.

5.1 Technical Data and Study Needs (Both Workgroups)

1. More conclusive data are needed on the desirable chemical form of DU (oxides and metal), the associated leach characteristics compared to spent fuel, and its influence on the geochemistry effects to retard radionuclide dissolution and diffusion away from the waste form. In particular, affecting the rate of movement away from the waste form for U and Tc are of interest.

2. Performance assessments that show the impact of inclusion of DU in the repository are needed to confirm that it can be emplaced without detrimental effects and to show likely enhancement effects on migration of radionuclides.

3. The physical and chemical properties of DUCRETE need to be better known. Needs include: DUCRETE thermal conductivity and the capability of DUCRETE to tolerate high temperatures comparable to those tolerated by conventional concrete for long time periods.3

4. Performance of DU shielded cask, e.g., DUCRETE, needs to be established by test demonstration of construction and performance data with an actual cask or a scaled model of an actual cask.

6. RECOMMENDATIONS FROM THE WORKSHOP

Participants gained a broader understanding of the issues and opportunities associated with the use of DU in the repository. The workshop provided a forum for improving communications on this subject between different parts of DOE. The workshop presented the potential for organizing some continuing periodic workgroup interaction and joint effort among the RW/YSMO, EM, and NE interests. Participants agreed that studies should be performed in the three areas mentioned previously, and that the workgroup participants should develop a draft list of such studies for consideration by the DOE Program Managers.

Although considerable information has been acquired on DU use, further technical and economic information is needed to assess whether use of DU should be seriously considered. Some potential contributions and value exist, but economic analysis from a total DOE system-wide perspective are needed before specific uses can be seriously considered. Cost information is needed, ranging from

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3 Testing on DUCRETE concrete at INEEL since the workshop has produced results in 28-day exposure tests showing that DUCRETE concrete and regular concrete behave similarly at 250°C, P. A. Lessing, Lockheed Martin Idaho Technologies Corporation, October 1997.
conversion of uranium hexafluoride to interim storage of spent fuel or high-level waste, to useful benefits in the mined geologic repository compared to alternative management and disposition options. Selected technical information should be acquired to support the economic analyses.

1. Studies, if favorable, should progress toward a position paper that can be presented to DOE senior management concerning the use of DU. Future clarification of the geochemistry and controlled migration functions is needed particularly for the emplacement of HEU spent fuel. Data are needed to calculate the effects of DU on the performance assessment throughout the performance period. Heat transfer and temperature calculations require data on thermal performance of various DU materials.

2. Backfill concepts, whether internal or external to the waste package, were not considered favorably because of anticipated poor low heat transfer associated with internal backfill, given the peak cladding temperature requirement. The strategy for drift backfill, even with non-DU materials, is a possible subject of future study since backfill will not occur for 50-100 years, if at all.

7. WORK SCOPE FOR WORKSHOP FOLLOW-ON EVALUATIONS

The following options were selected by the workshop attendees for further evaluation to determine the comparative benefits and costs of using depleted uranium in selected repository design modifications. These items are anticipated to be the responsibility of the YMSCO and its management and operations (M&O) contractors.

All cost impacts should use the assumption that the depleted uranium aggregate is provided as a government furnished material and should not assume that conversion cost burden from UF₆ to aggregate will accrue to the repository program. Fabrication of components may or may not accrue to the waste fund and as such should be accounted for as a separate line item.

7.1 Evaluation of Shield Casks

The major tasks that require further evaluation are presented in Sections 7.1.1 through 7.1.12 below.

7.1.1 DUCRETE Cask Conceptual Design and Cost Estimate For Horizontal Emplacement with Natural Circulation Cooling

This task will develop a conceptual design and include calculations to determine the effectiveness of natural circulation cooling in the annular geometry of a cask surrounding the waste package in the horizontal orientation. This natural circulation mode of heat rejection is critical to maintaining fuel temperature at or below 350°C. Cask vent modifications designs will be evaluated to provide suitable cooling pathways. Shielding analysis will be performed to evaluate the dose impact associated with various vent designs required for cooling. DUCRETE concrete composition should be optimized based upon the expected radiation field characteristics -- balancing the properties of DUCRETE such that
gamma and neutron radiation fields are more or less equally shielded. This will be dependent on the relative field strength of the relative radiation sources as well as the contribution of secondary gamma radiation induced from neutron attenuation. Wall thickness variations will be evaluated to determine the dose impact of the Multi-Use versus Repository Shield DUCRETE Cask Options.

7.1.2 DU Metal Cask Conceptual Design and Cost Estimate For Horizontal Emplacement

This task will develop a conceptual design and include calculations to determine the thermal performance of a DU metal cask to maintain fuel temperature at or below 350°C. Neutron and gamma shielding calculations will be performed to assess cask shielding effectiveness as a function of total package mass. Trade-off studies will be performed to assess the merits of placing the DU metal cask inside of the repository waste package versus placing the waste package inside of an oversized cask. Wall thickness variations will be evaluated to determine the dose impact of the Multi-Use versus Repository Shield Cask Options.

7.1.3 Repository Ventilation Requirements and Cost Impact Estimate

This task will evaluate the impact upon repository ventilation system design associated with maintaining the peak drift exit temperature at 40°C for an extended period of 100 to 200 years to support the use of DUCRETE casks. Fuel centerline temperature for this system will also be evaluated. The evaluations will be performed to determine increased circulation and ventilation drift requirements and associated capital and operating costs. The calculations should consider the impact of providing moisture augmentation to the airflow to increase the specific heat of the air stream and the associated heat removal as a function of total system flow rate. The cost impact of the augmented moisture requirements should also be evaluated. A moisture upper limit of 60% relative humidity is assumed acceptable for repository operation but should be confirmed.

7.1.4 Repository Emplacement System Modifications and Cost Impact

The shielded cask will increase the weight of the emplaced disposal package to weights between 90 to 140 tons, depending upon the detailed design chosen (Multi-Use versus Repository Shield DUCRETE Cask Options). This task should evaluate the design modifications resulting from the replacement of a shielded waste package on the rail transport and gantry emplacement system. An alternative emplacement system could be considered if appropriate. Any repository emplacement system design simplifications identified associated with the elimination of remote handling features should be clearly identified and cost impacts estimated. The intangible value of system maintainability in a contact-handled environment should be parametrically evaluated as a function of postulated remote system failures.

7.1.5 Surface Facility Modifications and Cost Impact

This task should evaluate the cost of handling a heavier shielded waste package in the fuel handling facility loading areas. Any reduction of facility shielding requirements should be included as
appropriate. If surface facility designs could take advantage of the lag storage potential intrinsic to Multi-Use shielded casks, this option should be evaluated and the benefits quantified.

7.1.6 Long-Term Inspection and Maintenance of Underground Facilities

A study should be conducted to estimate the long-term impact of inspection and maintenance activities during the pre-closure period. Estimates should include the lower cost of sensors and inspection equipment qualified for operation at 40°C or below and at lower maximum ambient humidity. This study should also evaluate the inspection options associated with using vent holes or other penetrations in a cask for insertion of waste package inspection equipment.

7.1.7 Licensability and Schedule Impact

An evaluation of potential licensing issues should be conducted. Potential simplifications associated with the use of shielded waste packages in repository operations should also be evaluated. An evaluation of the potential repository schedule impact associated with DU cask deployment should be made. Any associated impact on the receipt of fuel based on using the casks as lag storage should also be made. Current regulatory limitations on interim storage at the repository impeding the use of lag storage should be identified.

7.1.8 Performance Assessment Impact

The impact of the additional heavy metal source term should be evaluated assuming the uranium is present as 1) a stable ceramic aggregate in concrete, and 2) a metal. The assessments should consider the potential benefits of the cask as a drip shield, as a source of U-238 for dilution of fissile materials and as a retardant for groundwater dissolution of the fuel due to uranium saturation effects on the groundwater. Thermal analysis of the waste package in the post-closure environment should be evaluated to determine if there are any unforeseen barriers to maintaining the fuel thermal limit of 350°C. Any reduction of waste package corrosion associated with the reduced temperature and humidity environment during the ventilation period should be quantified.

7.1.9 Comparative Evaluation of Pre-Closure Monitoring

The determination of waste package performance during pre-closure is an important function that must be performed as required in 10 CFR 60. Results of monitoring of in-situ waste packages will be used to determine the long-term performance and will also serve as the basis for backfill and final closure decisions. Waste package performance is also a major element in the current robotics program for the repository. Parameters likely to be monitored will include radiation field measurements, local [to each package] airborne radiation leakage, corrosion status, and emplacement drift roof support structure integrity, to name a few. The use of a self-shielded waste package offers an opportunity to perform such monitoring in a direct manner as compared to the use of a robotic inspection gantry system. Use of shielded packages, on the other hand, introduces a number of waste package inspection complexities that must also be evaluated.
7.1.10 Pre-Closure Operations Health and Safety Impact

An analysis of the expected impact on personnel dose and occupational risks associated with remote versus contact handled operations at low temperature should be developed.

7.1.11 Closure Related Activities and Cost Estimate

An estimate of the impact on closure activities, e.g., backfill, should be made. Any other closure activities that are affected by the radiation environment should be identified and cost impacts estimated.

7.1.12 Delayed Closure and Total Life Cycle Cost Impact

The cost of extended ventilation system operations for an additional 100 to 200 years after the emplacement activities should be evaluated. The impact of all positive and negative cost impacts should be compiled to assess the cost to the waste fund for this option.

7.2 Evaluation of Half-Cylindrical Shield With End Ventilation

The major tasks that require further evaluation are presented in Sections 7.2.1 through 7.2.10 below.

7.2.1 Half-Cylindrical Shield Conceptual Design and Cost Estimate

This task will include calculations to determine the effectiveness of using a half-cylindrical shield surrounding the waste package. Shielding analysis will be performed to evaluate the dose impact associated with various designs as a function of wall thickness and length. DUCRETE concrete composition should be optimized based upon the expected radiation field characteristics -- balancing the properties of DUCRETE such that gamma and neutron radiation fields are more or less equally shielded. This will be dependent on the relative field strength of the relative radiation sources as well as the contribution of secondary gamma radiation induced from neutron attenuation.

7.2.2 Repository Ventilation Requirements and Cost Impact Estimate

This task will evaluate the impact upon repository ventilation system design associated with maintaining the peak drift exit temperature at 40°C for an extended period of 100 to 200 years. Fuel centerline temperature for this system will also be evaluated. The evaluations will be performed to determine increased circulation and ventilation drift requirements and associated capital and operating costs. The calculations should consider the impact of providing moisture augmentation to the airflow to increase the specific heat of the air stream. The cost impact of the augmented moisture requirements should also be evaluated. A moisture upper limit of 60% relative humidity is assumed acceptable for repository operation but should be confirmed.
7.2.3 Repository Emplacement System Modifications and Cost Impact

The shielded half cylinder is assumed to be emplaced with the waste package. This task should evaluate the design modifications resulting from the replacement of a shielded waste package on the rail transport and gantry emplacement system. An alternative emplacement system concept could be considered if appropriate. Any repository emplacement system design simplifications identified associated with the elimination of remote handling features should be clearly identified and cost impacts estimated. The intangible value of system maintainability in a contact handled environment should be parametrically evaluated as a function of postulated remote system failures.

7.2.4 Surface Facility Modifications and Cost Impact

This task should evaluate the cost of handling a heavier shielded waste package in the fuel handling facility loading areas. Any reduction of facility shielding should be included as appropriate.

7.2.5 Long-Term Inspection and Maintenance of Underground Facilities

A study should be conducted to estimate the long-term impact of inspection and maintenance activities during the pre-closure period. Estimates should include the lower cost of sensors and inspection equipment qualified for operation at 40°C or below and at lower maximum ambient humidity. This study should also evaluate the inspection options associated with using open ends or other penetrations in the shield for insertion of waste package inspection equipment.

7.2.6 Licensability and Schedule Impact

An evaluation of potential licensing issues should be conducted. Potential simplifications associated with the use of partially shielded waste packages in repository operations should also be evaluated. An evaluation of the potential repository schedule impact associated with half-cylinder deployment should be made.

7.2.7 Performance Assessment Impact

The impact of the additional heavy metal source term should be evaluated assuming the uranium is present as a stable aggregate in concrete. The assessment should consider the potential benefits of the half-cylinder as a drip shield, as a source of U-238 for dilution of fissile materials, and as a retardant for groundwater dissolution of the fuel due to uranium saturation effects on the groundwater. Thermal analysis of the waste package in the post-closure environment should be evaluated to determine if there are any unforeseen barriers to maintaining the fuel thermal limit of 350°C.

7.2.8 Pre-Closure Operations Health and Safety Impact

An analysis of the expected impact on personnel dose and occupational risks associated with remote versus contact handled operations at low temperature should be developed.
7.2.9 Comparative Evaluation of Pre-Closure Monitoring

The determination of waste package performance during pre-closure is an important function that must be performed as required in 10 CFR 60. Results of monitoring of in-situ waste packages will be used to determine the long-term performance and will also serve as the basis for backfill and final closure decisions. Waste package performance is also a major element in the current robotics program for the repository. Parameters likely to be monitored will include radiation field measurements, local [to each package] airborne radiation leakage, corrosion status, and emplacement drift roof support structure integrity, to name a few. The use of a self-shielded waste package offers an opportunity to perform such monitoring in a direct manner as compared to the use of a robotic inspection gantry system. Use of shielded packages, on the other hand, introduces a number of inspection complexities that must be evaluated.

7.2.10 Delayed Closure and Total life Cycle Cost Impact

The cost of extended ventilation system operations for an additional 100 to 200 years after the emplacement activities should be evaluated. The impact of all positive and negative cost impacts should be compiled to assess the cost to the waste fund of this option.

7.3 Evaluation of Emplacement of Depleted Uranium Aggregate in Inverts on Drift Floor Under Waste Packages

The major tasks that require further evaluation are presented below in Sections 7.3.1 through 7.3.6.

7.3.1 DU Invert Conceptual Design

This task will include calculations to determine the detailed invert design including weight, uranium aggregate content and other related information for inverts used in the drifts. The total number of inverts and the associated quantity of depleted uranium aggregate should be estimated.

7.3.2 Repository Emplacement System Modifications and Cost Impact

The DUCRETE inverts will be heavier than conventional concrete and the impact upon emplacement should be estimated.

7.3.3 Surface Facility Fabrication and Cost Impact

This task should evaluate the cost of casting DUCRETE inverts at a surface facility designed to prevent spread of depleted uranium oxide dust. Any radiological impact during manufacturing should be identified.
7.3.4 Licensability and Schedule Impact

An estimate of potential licensing issues and simplifications associated with the use of DUCRETE inverts should be made.

7.3.5 Performance Assessment Impact

The impact of the additional heavy metal source term should be evaluated assuming the uranium is present as a stable aggregate in concrete. The assessment should consider the potential benefits of the invert as a source of U-238 for dilution of fissile materials and as a retardant for groundwater dissolution of the fuel due to uranium saturation effects on the groundwater.

7.3.6 Total Life Cycle Cost Impact

The cost of DUCRETE inverts compared to conventional concrete inverts should be identified as well as any other costs which significantly impact total life cycle cost.

8. ADDITIONAL BACKGROUND INFORMATION

The Workshop attendees were provided additional background information on depleted uranium uses and prior studies before and at the Workshop. This material was factored into the deliberations and is included in Appendix J to complete the record of this workshop.

9. REFERENCES

APPENDIX A

Workshop Attendees

1. Carl Cooley, DOE HQ, EM-50
2. Paul Harrington, DOE RW, YMSCO
3. Martin Haas, M&O, MGDS
4. Paul DeLozier, consultant to EM-50
5. Bob Burgoyne, Booze Allen
6. Allen Kyes, INEEL-NSNFP
7. Jim Dubrin, LLNL
8. Bob Barton, YMSCO
9. Steve Gomberg, DOE HQ, RW-51
10. Tim Gunter, DOE-SR
11. Allen Croff, ORNL
12. Steve Baker, consultant
13. Robert Heard, INEEL
15. Michael Taylor, EMEF, Oak Ridge
17. Stephen Leedom, DOE NV
18. Emile Bernard, SNL Germantown
19. Paul Daniel, Booze Allen/ymSCO
20. Paige Russell, DOE-RW, YMSCO
22. Abe Van Luik, DOE-RW, YMSCO
23. KAL Bhattacharyya, M&O, Repository
24. Dave Haught, DOE-RW, YMSCO
26. John Clouet, MGDS PEO/Waste Integration
27. Carl Detrick, NR-Bettis
29. Dan McKenzie, M&O, Repository Design
30. Ched Bradley, DOE HQ, NE-40
31. Collette Brown, DOE HQ, NE-40
32. M. Jonathan Haire, ORNL
33. Bill Quapp, Nuclear Metals, Inc.
34. Charlotte Johnson, SAIC, Germantown
35. Melda Rafferty, DOE-Ports.
36. Daniel B. Bullen, NWTRB
37. Carl Di Bella, NWTRB, staff
38. Gary Knight, Waste Policy Institute (proceedings editor)
39. Robert Mussler, Checkpoint Analysis (meeting facilitator)
APPENDIX B

Objective

WORKSHOP ON USES OF DEPLETED URANIUM IN STORAGE, TRANSPORTATION, AND REPOSITORY FACILITIES

The workshop will be organized into two workgroups: Workgroup A will cover all subsurface operations. Workgroup B will cover all surface operations including surface storage and transportation interfaces.

The objectives for each workgroup are:

1. Prepare descriptions of options/concepts and identify the future work to support decisions on use of depleted uranium;

2. Identify existing or new requirements that must be met or developed for specific applications or as an integrated system that functions for one or more of the three general applications (away from repository, repository surface, repository subsurface); and

3. Provide recommendations concerning the use of DU.

NOTE: During discussions at the workshop after the first morning’s presentations, the attendees decided to change the focus of the workgroups to examine pre-closure and post-closure issues. Therefore, the workgroups were accordingly renamed.
# APPENDIX C

## Agenda

**WORKSHOP ON USES OF DEPLETED URANIUM IN STORAGE, TRANSPORTATION, AND REPOSITORY FACILITIES**

### First Day July 15, 1997

<table>
<thead>
<tr>
<th>Time</th>
<th>Session</th>
<th>Presenter</th>
</tr>
</thead>
<tbody>
<tr>
<td>8:00</td>
<td>RW introduction&lt;br&gt;Welcome to YMPO, discuss status of project</td>
<td>Barton</td>
</tr>
<tr>
<td>8:15</td>
<td>Workshop objectives, product &amp; ground rules --Agreement on agenda</td>
<td>Facilitator</td>
</tr>
<tr>
<td>8:30</td>
<td>Overview of NE program to manage DU --Objectives/needs --Status of depleted uranium resource</td>
<td>Bradley</td>
</tr>
<tr>
<td></td>
<td>- locations/owners&lt;br&gt;- current quantities&lt;br&gt;- projected quantities&lt;br&gt;- assay&lt;br&gt;- DOE obligations&lt;br&gt;- NE program issues</td>
<td></td>
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<tr>
<td>9:15</td>
<td>Overview of EM studies on DU options --Objectives and needs --Status of depleted uranium studies</td>
<td>Cooley</td>
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<tr>
<td></td>
<td>- Options studied&lt;br&gt;- Quantities used&lt;br&gt;- EM quantities &amp; locations/owners&lt;br&gt;- Assay&lt;br&gt;- DOE obligations&lt;br&gt;- EM program issues</td>
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<tr>
<td>10:00</td>
<td>Break</td>
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<tr>
<td>10:15</td>
<td>Overview of MGDS design/operations --Reference repository design --Waste package details</td>
<td>Harrington</td>
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C-1
Agenda (Continued)

WORKSHOP ON USES OF DEPLETED URANIUM IN STORAGE, TRANSPORTATION, AND REPOSITORY FACILITIES

--Performance requirements for DU in WP/EBS and underground facilities at Yucca Mountain
--Potential applications considered (inverts, lining, WP fill, WP corrosion allowance barrier, drip shields, backfill, shield doors)

11:00 Overview of Waste Acceptance Storage Transportation (WAST) design/operations

--Summary of potential CISF: Tasks now and
--DTS design
--Operations
  - Potential applications and performance requirements for DU at hypothetical ISF, surface facilities, and within OCRWM
--Transportation system
  - Potential applications (transportation casks, hot cells)
  - Performance requirements (thermal, shielding, mechanical, criticality, corrosion, spatial)
--Dry storage cask requirements

12:00 Lunch

1:00 DU Options for potential use in the repository for working group consideration

--Repository surface (handling, storage, & transportation
--Repository subsurface (shielding, construction, & waste package)

2:00 Depleted Uranium Silicate Container Backfill System (DUSCOBS) and other ORNL backfill studies

2:30 DU backfill studies for the YM Project

Bob Burgoyne, BAH

SystemTeam (Quapp et al)

Croff/Haire

W. Clark(LLNL)
Agenda (Continued)

WORKSHOP ON USES OF DEPLETED URANIUM IN STORAGE, TRANSPORTATION, AND REPOSITORY FACILITIES

3:00 DUAGG and DUCRETE property summary
- chemistry
- neutron/gamma shielding
- corrosion properties
- thermal properties
- mechanical properties
- fabrication experience
- schedule requirements

R. Heard

3:30 Break

3:45 Workshop group discussions for breakouts
- Group issue development
- Approach and products expected
- Report outline consensus
- Working group participants
- Select group leaders

Facilitator & WG Chairs

5:15 Adjourn

Second Day, July 16, 1997

8:00 Review of charge of working groups
 Facilitator

8:15 Convene two simultaneous working groups on subsystems

Group 1 -- Repository site surface handling subsystems
Group 2 -- Repository site subsurface including long-term issues

Identify interface and integration issues and prepare input to report, include minimum quantify for benefit, maximum useable quantity, unit or total cost impact
Agenda (Continued)

WORKSHOP ON USES OF DEPLETED URANIUM IN STORAGE, TRANSPORTATION, AND REPOSITORY FACILITIES

2:00  Working groups identify issues relevant to technical information needs for use

4:00  Joint Working Group summarization of issues and follow-up work that needs to be done to support the scenarios, etc.

5:00  Adjourn for drafting of report sections

Third Day, July 17, 1997

8:00  Joint meeting of working groups

9:00  Working groups complete draft

11:30 Follow-up action requirements to finish report

12:30 Working group submissions completed

1:15  Chairmen of each working group edit draft of report and send to the group for comments. Decide on actions to complete the report draft to NE/RW-YMPO/EM

4:00  Adjourn
APPENDIX D

Synopsis of Workshop Presentations

DOE staff from OCRWM, YMSCO, NE, EM, and support and technical contractors met in the Las Vegas offices of the YM M&O July 15-17, 1997, to discuss various options for using DU in the geologic repository, to make recommendations concerning which options seemed most viable, and to list what information needs would be required for DOE to address such possibilities. Robert Mussler was the facilitator for the meeting. Where available, copies of the overhead slides from the presentations summarized below are found in Appendix E.

Presentations

The first speaker was Bob Barton, YMSCO, who welcomed the group to Las Vegas and detailed for the attendees the major program milestones that are coming up: finalization of the 10 CFR 960 rule, viability assessment, publication of a draft EIS, publication of the final EIS and Record of Decision (ROD) in the year 2000, submit site recommendation to the President in 2001, and submission of the licensing application to NRC in 2002. For the Viability Assessment (VA) required in 1998, the project will develop a preliminary design concept of the repository using a systems engineering approach.

The repository is being designed to hold 70,000 MTU of commercial SNF. Ten percent equivalent (7,000 MTU) of the stored material will be from DOE (from production reactors, research fuel, navy fuel, civilian HLW, defense HLW, and surplus plutonium). Additional work will be needed after the VA. Completion of the Licensing Application (LA) design will include design options for defense-in-depth purposes. To support the VA in 1998 another total system performance assessment (TSPA) will be performed.

For preparation and issuance of a final EIS, compliance must be demonstrated with 10 CFR 960 and the DOE siting guidelines. DOE has had preliminary discussions with NRC on their technical work. EPA has not issued the standards toward which the project must comply, so DOE is working using interim guidelines. They see no “show stoppers” and are confident in their ability to produce the VA product. The NRC will have to reissue 10 CFR 60 depending on the new EPA standards. EPA has not committed to a schedule for issuance and is a year a half late. The 10 CFR 960 standards are siting guidelines and looks at the total systems rather than subsystems (Total System Performance Standards).

Steve Gomberg, DOE HQ, said that the purpose of this workshop is to evaluate the feasibility of using depleted uranium to assist OCRWM in meeting the performance objectives of the civilian radioactive waste management system. No decisions are planned to be made during this workshop. Final decisions would be made based on numerous considerations, including technical, cost, and other tradeoffs.
This workshop will focus on uses of DU in the surface and subsurface facilities at a potential Yucca Mountain repository. Significant information is already available for using depleted uranium as part of transportation and storage system designs, primarily for radiation shielding. Numerous technical reports have been prepared which look at benefits of depleted uranium in nuclear and other applications. Several existing technologies using depleted uranium metal have been licensed and fabricated.

There is a National focus on the proper management of the country’s supply of depleted uranium. It is currently stored in metal cylinders as uranium hexafluoride or as an oxide. The Office of Nuclear Energy within DOE recently released a draft EIS on depleted uranium. The DOE must recommend safe and cost effective solutions for the conversion and storage of depleted uranium. It must also assess the potential uses of the Federal Government’s surplus materials for reuse.

The Office of Civilian Radioactive Waste Management is presently evaluating the disposal of materials in the first geologic repository. This being conducted under the Nuclear Waste Policy Act and the National Environmental Policy Act. For this meeting, we are not evaluating the disposal of DU in a geologic repository. We are evaluating the possible uses of DU as a material of construction. This could include as a metal, and oxide, an aggregate, or a silicate. However, one thing is clear: this is an opportunity to integrate activities of National concern and priority. It is hoped that many offices within the Department can gain technically, economically, and gain the support of necessary constituencies and stakeholders beneficial to RW’s, Em’s, and NE’s missions.

This workshop includes the participation of experts who can identify possible uses of DU assess its relevant properties as applied to the mined geologic disposal system, and identify additional information to make recommendations to DOE to support future decisions. It is expected that a report documenting the deliberations resulting from this workshop will be prepared.

Charles Bradley, NE, DOE-HQ, then gave a report on the management of the DU management program. The UF₆ inventory of 560,000 MT is contained in 46,422 cylinders which constitutes about one-half of all the uranium mined in the entire world (another 30-40% of which is in Russia). They are painting all the cylinders now, which are 62% full of solid UF₆. The U.S. Enrichment Corporation is filling an additional 1500-2000 cylinder/year. Until privatization of the USEC occurs, DOE is responsible for managing all of the additional generated DUF₆. The assay of the UF₆ is 272,000 MT of < .3% U-235; 260,000 MT of .3-.41%; and 28,000 MT of .41-.71%.

The main purpose of the $10 million/year NE DU management program is to protect cylinders that are currently lying on the ground. They expect to convert all the UF₆ to an oxide by 2020. The State of Ohio began seeking management control of the material under RCRA in late 1987. DOE recently reached a settlement with the State of Ohio (with the legal agreement being signed imminently) to avoid declaring depleted uranium a waste. Safety improvements, required by DNFSB finding 95-1, are underway. The DU is classified as a “source material” under the Atomic Energy Act of 1954, but if it is called a waste, then DOE has a real problem. The settlement agreement with the State of Ohio will legally determine it to be a non-problem for ten years.
NE plans to finish the cylinder and yard improvements and to resolve the Ohio notice of violations at the Portsmouth facility. They will complete the PEIS and implement the Record of Decision (ROD) probably in mid-1998. The implementation of their management program will involve two phases. Phase I is strategy selection (an engineering analysis report and a cost analysis report by LLNL and finalization of the PEIS by ANL). Phase II is implementation of the ROD incorporating the selected long-term management strategy. They presently plan to begin conversion of the UF₆ to a more stable oxide in 2005, finishing in 2020. Present preliminary costs are: conversion to oxide $2-6/kg UF₆; conversion to metal $5-12/kg UF₆; disposal of oxide $1/kg UF₆; transportation $~1/kg UF₆; and use as oxide or metal $6-10/kg UF₆.

Issues to resolve are 1) any processing or transportation issues, 2) NE cannot expect a big increase in its budget, and 3) industry capital investment will be required to process or convert and this is expensive.

Conclusions NE has made are: 1) DU and fluorine (180,000 MT included in the resource) are DOE assets, 2) they expect more than one use to be pursued, and 3) they must work with industry to develop mutually agreeable arrangements.

Carl Cooley, EM-50 at DOE HQ then presented the EM perspective. The EM effort began in FY 92 at the same time that Ohio’s concerns came forward. Since EM felt it did not need any more disposal business, it began studying the use of DU for alternative uses. It tried to identify areas where R&D efforts might be needed to either develop more efficient conversion technologies or manufacturing processes. He showed a series of slides laying out EM’s preliminary estimates for both conversion (to either oxide or metal) contrasted with the cost of disposal of the resource at different sites either as a LLW or as a RCRA waste. He detailed how INEEL, in studying the problem, invented a new material called DUCRETE concrete which uses DU-aggregate (DUAGG) to make a concrete with enhanced radiation attenuation capabilities compared to conventional concrete. Conceptual designs of DUCRETE dry fuel storage casks identified both size and weight reduction compared to the traditional designs.

EM’s contractors are now producing test quantities of DUAGG, at a pilot scale aggregate production facility. The requirements for these mass-produced DUAGG briquettes are based on bench scale, hand-pressed DUAGG specimens. They will cast this DUCRETE into forms and test its performance. They will optimize the DUAGG quality, confirm the chemical stability at elevated temperature, and decide on applications to develop performance specifications. Tests will be performed to confirm theoretical neutron and gamma attenuation performance. They also need to determine how to get industry participation in developing DUCRETE applications.

Paul Harrington, the team leader for Viability Assessment at YMSCO, then talked about present repository design plans citing the newly published “Repository Reference Design Description” document, which lays out the design for 100 year surveillance prior to drift closure and fuel recovery up to that time, if desired. DOE RW wants to change its philosophy from “bury and forget” to “indefinite storage”, meaning monitoring the stored material by ensuring the integrity of the package in
terms of thermal, radiation, and drift stability. 10 CFR 60 requires 50 years of monitoring; YMSCO would like to change that to 100 years. Congress will determine which of the five proposed new rail routes to the mountain will be used for transport.

He briefly described the present repository design. The repository consists of approximately 200 18-foot diameter emplacement drifts. The main transport drifts are 25 foot diameter and lined with either precast or cast-in-place drift liners. There are questions whether concrete is maintenance free for 100 years in a temperature environment up to 200°C. The emplacement drifts are roughly one mile long (with a ventilation shaft cutting each roughly in half) and are located 100 meters above the water table. The emplacement drifts are traversed by gantries on a 3-foot gauge rail for emplacement of the waste packages into the drift. Waste packages weigh up to 70 MT. After emplacement, the waste packages must be monitored for radiation leaks. Leakers can be removed by the gantry which can moved over emplaced packages, enabling removal without having to pull out all the other packages outboard of the leaker.

The drift design will inter mix different waste packages including PWR, BWR, DOE HLW, and defense SNF to adjust the total thermal load in a drift. DOE highly enriched SNF will be loaded inside of DOE HLW packages to ease criticality concerns. There are several thermal limits: 350°C for the fuel cladding temperature, 200°C on the drift wall temperature, and 90°C on the zeolite mineral layer below the repository level.

Harrington then discussed the required Part 60 processing for material acceptance in the repository. The material cannot be pyrophoric, reactive, or unstable so the UF₆ must be made more stable and converted into one of the following forms: 1) an oxide, carbide, or silicate; 2) UF₄, a non-volatile solid, must be processed to remove the fluoride for repository disposal; 3) UO₂ must be processed into a non-dusting form for repository use; and 4) DU metal may need to be formed into shapes compatible with waste package designs.

He then challenged the group to consider all of the benefits and impacts of any of the DU use options when the breakout sessions were held. It was further recognized that it may eventually be concluded that there would be no economically justifiable use for DU in the repository even though technical arguments for some benefits might exist.

Bob Burgoyne of Booze, Allen & Hamilton, supporting contractor for the Office of Waste Acceptance, Storage, and Transportation (WAST) at RW then detailed the WAST program. They have just produced a Topical Safety Analysis Report (TSAR) for the Central Interim Storage Facility (CISF) to hold 40,000 MTU to be done in phases. Phase I would take canistered spent fuel in rail cars, and Phase II would involve a mix of canisters and bare fuel. The purpose of the TSAR is to get a dialogue going with the NRC and to get issues identified so work can commence on them. The submittal of the CISF TSAR enhances the success of CISF deployment by being non-site specific and reducing the direction of the NRC's review. Development of the market driven transportation acquisition approach will mean DOE is no longer designing casks but relying on industry. This will
reduce lead time, identify key industry and institutional issues, and target development of industry and institutional systems.

A status of the market-driven program included two pre-solicitation meetings and a draft RFP which was issued for comments in December 1996. A revised draft is scheduled for release in late 1997. A final RFP will be released pending completion of site viability.

Regional Servicing Contractor(s)⁴ (RSC) will provide acceptance, transportation and delivery services for commercial Spent Nuclear Fuel (SNF) in three distinct contract phases. Phase A will focus on developing the detailed planning necessary for performance of Phase B mobilization and Phase C operations activities. Additionally, during Phase A, fixed prices/rates for equipment acquisition and for servicing the owners and generators (Purchasers) of SNF under the Standard Contract will be developed.

Bill Quapp of Nuclear Metals, Inc. (and formerly the DU Principal Investigator for EM-50 at INEEL) then gave a presentation entitled “Engineering Considerations for the Use of DUCRETE Shielded Waste Packages for SNF and HLW”. The presentation provided a description of a DUCRETE cask for shielding the disposal waste package in several different disposal scenarios. These included vertical and horizontal emplacement in drifts and a concept where a large room rather than a narrow drift was mined to allow a cluster of shielded waste packages in one location. Instead of an 25 foot diameter shaft in the repository, he proposed as one option a drift cross-section a half-circle 60' wide and 30' tall. He then discussed in brief the sixteen options that he and the other two consultants engaged by EM-50 had prepared to identify possible uses of DU in the repository program. These concepts are not repeated here but are covered in detail in the workgroup reports in Appendices E, G and H.

Allen Croff of the Oak Ridge National Laboratory presented a paper on “DU Oxide Fill for Fuel Packages”, a concept developed originally by Dr. Richard Forsberg at ORNL for plutonium, now modified for DU. Beads of DU (called Depleted Uranium Silicate Container Backfill System or DUSCOBS) are not spherical nor uniform by design. Three metric tons of DU could be used as glass filler beads in a 21 PWR assembly wherein 70% of the void space is filled. If you use three different particle sizes, you can get a 95% void fill rate. Helium is the reference fill gas for these waste packages. They have not considered using uranium metal shot for this purpose.

Bob Heard of INEEL spoke on DUAGG and DUCRETE and elaborated in detail on the process to develop each concept. He said the density of the oxide with which you start does not really matter. There is no detectable leaching from the aggregate (DUAGG) in tests using the ANS 16.1 procedure. DUAGG with a density of about 8.5 gm/cm³ and a porosity of less than 2% has been made in bench scale tests. In FY 96, development concentrated on aggregate composition. In FY 97, development concentrated on aggregate production. They are now planning on doing additional tests: DUAGG oxidation testing, DUCRETE strength testing, and DUCRETE shielding testing. DUAGG composition

⁴ The term RSA (Regional Servicing Agent) used in the previous draft RFP has been changed to RSC (Regional Servicing Contractor) because an RSC will perform as a DOE contractor and this term more accurately describes its status. The responsibilities described previously for the RSA now apply to the RSC.
is 80% by volume UO₂ and 20% additives (soil, clay, mill additives, and boron). No testing has been
done on the thermal conductivity properties of DUCRETE. This is needed for DUCRETE in cask
design.

**General Discussion**

Carl Cooley, EM-50, again thanked the presenters for getting everyone up to speed and reiterated that
the mission of the workshop was to look at the state of knowledge with respect to DU and to take a step
back and look at it from a broad systems viewpoint. The key now is how to put it together over the next
couple of days into a coherent meeting report. He said there were possibilities for putting something
good together because we have the expertise here to flesh out any options the group might want to
recommend. It should not be looked at in terms of whether EM or NE or RW should do something;
only whether something should be done to resolve issues surrounding a DU use concept, if that is the
workshop’s consensus. And a “no role for DU” is OK, too, but what we have to address, and he said he
is queried about it all the time at HQ, is there a better outcome that can be achieved if DU is used for
“x” purpose. Of course, this assumes the regulatory picture is conducive.

After the facilitator reiterated the agenda for the last two and a half days of the workshop, the group
came up with a format against which to weigh each of the possible DU use options:

1) Identify a range of potential applications

2) Identify benefits/drawbacks of each option

3) Identify obstacles/uncertainties associated with implementing the option

4) Regulatory issues raised by the option

5) What needs to be done to address/resolve the obstacles/uncertainties of each option

Paul Harrington, DOE-YMSCO, summarized what he thought might be the possible benefits of using
DU in the YMSCO program from a performance requirements point of view.

1) Criticality control both internal to the package (fills) and external (backfill)

2) Shielding, though with a series of constraints: thermal, package weight, and envelope size

He suggested for each option, the costs to RW, EM, and NE should be identified which would add
some options and cause others to be avoided. The availability of material versus the date needed by the
program might also impact the choice of one or more options. The question needs to be discussed
regarding who is going to pay for the costs to check out potential benefits. Other topics that should be
discussed were repository performance, R&D work required, and basic performance assessment criteria.
Finally, the group decided to scrap the workgroup break out of "above ground" and "below ground" options as constructed by the workshop's planners. The agreed upon split was to discuss each option as to whether it was a benefit from either a "pre-closure" or a "post-closure" point of view and the workgroups were renamed accordingly. The attendees were then assigned to a workgroup with a fairly even number of people representing RW, NE, and EM in each group.
### APPENDIX E

#### Presentation Materials

<table>
<thead>
<tr>
<th>Title of Presentation</th>
<th>Presenter</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Welcome to the Yucca Mountain Project</td>
<td>Paul Harrington</td>
</tr>
<tr>
<td>2. Depleted Uranium Hexafluoride Management Program</td>
<td>Charles E. Bradley</td>
</tr>
<tr>
<td>3. EM OST Program Introductory Remarks and Program Overview</td>
<td>Carl R. Cooley</td>
</tr>
<tr>
<td>5. Engineering Considerations for the Use of DUCRETE Shielded Waste Packages for SNF and HLW</td>
<td>William Quapp, Paul DeLozier, &amp; Martin Haas</td>
</tr>
<tr>
<td>6. Depleted Uranium Oxide Fill for Fuel Packages</td>
<td>Allen G. Croff</td>
</tr>
<tr>
<td>7. DUAGG and DUCRETE Properties</td>
<td>Robert E. Heard</td>
</tr>
</tbody>
</table>
Welcome to the Yucca Mountain Project
Yucca Mountain Viability Assessment by 1998

“The completion of the constituent elements of the viability assessment constitute a logical convergence at which the Program can make a measurably improved appraisal of the prospects for geological disposal at the Yucca Mountain site. The assessment is an interim step in the process leading to a site recommendation to the President....”

From the Fiscal Year 1998 Budget of the United States Government Appendix, p. 475
## What are the Wastes that Must be Managed?

<table>
<thead>
<tr>
<th>Waste Type</th>
<th>Form</th>
<th>Quantity</th>
<th>Location</th>
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<tbody>
<tr>
<td>Spent Nuclear Fuel</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Commercial SNF</td>
<td>Reactor fuel assemblies</td>
<td>Current ~ 33,000 MTU</td>
<td>72 Operating and shut down reactor sites</td>
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<td></td>
<td></td>
<td>Projected ~ 85,000 MTU</td>
<td></td>
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<tr>
<td>DOE SNF</td>
<td></td>
<td>2,720 MTHM</td>
<td>Savannah River</td>
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<tr>
<td>Production Reactors</td>
<td>N and single pass reactor fuel</td>
<td>(2120)</td>
<td>Hanford</td>
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<tr>
<td>Research/Test Reactors</td>
<td>Shippingport, Pathfinder,</td>
<td>(115)</td>
<td>Idaho</td>
</tr>
<tr>
<td></td>
<td>Fermi, ATR, TRIGA</td>
<td></td>
<td>Other sites</td>
</tr>
<tr>
<td>Reactors Debris</td>
<td>Three Mile Island Core 2</td>
<td>(83)</td>
<td></td>
</tr>
<tr>
<td>Commercial materials</td>
<td>Reactor fuel assemblies</td>
<td>(99)</td>
<td></td>
</tr>
<tr>
<td>Navy Fuel</td>
<td>Reactor fuel assemblies</td>
<td>(65)</td>
<td></td>
</tr>
<tr>
<td>Miscellaneous</td>
<td>Metallic Sodium Bonded</td>
<td>(238)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Driver Fuel &amp; Target</td>
<td></td>
<td></td>
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<tr>
<td>Civilian HLW</td>
<td>Liquid (vitrified in canisters)</td>
<td>2,315 m³ ~2.3x10⁷ Curies</td>
<td>West Valley Demonstration Project (300 canisters)</td>
</tr>
<tr>
<td>Defense HLW</td>
<td>Liquid, Sludge (vitrified in canisters)</td>
<td>371,170 m³ ~7.68x10⁷ Curies</td>
<td>Hanford 12,200 canisters)</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>Savannah River (5717) canisters</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Idaho (704 canisters)</td>
</tr>
<tr>
<td>Surplus Plutonium</td>
<td>Metal (immobilized with HLW or converted to MOX fuel)</td>
<td>50 MT</td>
<td>Pantex, Hanford, SRS, LANL, ORR, RFETS</td>
</tr>
</tbody>
</table>
Additional Work Needed After VA

- Completion of LA design, including design options to be included for defense-in-depth

- Completion of confirmatory testing to enhance confidence in site models

- Incorporate enhanced understanding of site and design process models in TSPA-LA
Additional Work Needed After VA (cont’d)

- Preparation and issuance of the Final EIS
- Demonstration of compliance with 10 CFR 960, DOE siting guidelines
- Intensive interaction with NRC of our technical work
- Demonstrate compliance with evolving environmental standards
Summary

- Confident in our ability to produce the VA product, and finally

- No evidence of any show stopper in process to produce LA
DEPLETED URANIUM HEXAFLUORIDE MANAGEMENT PROGRAM

CHARLES E. BRADLEY, JR.

WORKSHOP ON POTENTIAL USES OF DEPLETED URANIUM IN A GEOLOGIC REPOSITORY

JULY 15 - 17, 1997
LAS VEGAS, NV
DEPLETED URANIUM HEXAFLUORIDE
MANAGEMENT PROGRAM

INVENTORY OF DEPLETED UF₆

• 560,000 METRIC TONS IN 46,422 CYLINDERS
• 28,351 CYLINDERS AT PADUCAH, KY
• 13,388 CYLINDERS AT PORTSMOUTH, OH
• 4,683 CYLINDERS AT OAK RIDGE, TN
• 95% OF ALL DOE DEPLETED URANIUM
DEPLETED URANIUM HEXAFLUORIDE
MANAGEMENT PROGRAM

OTHER DEPLETED URANIUM

- **UO₃** - 19.5 METRIC TONS - MOSTLY AT SAVANNAH RIVER
- **METAL** - 5.3 METRIC TONS - FERNALD & SAVANNAH RIVER
- **UF₄** - 3 METRIC TONS - PADOUCAH & FERNALD
- **OXIDES** - 145,000 KG - SAVANNAH RIVER
- **SCRAP** - 35,000 KG - SAVANNAH RIVER
- USEC IS PRODUCING 1500 TO 2000 CYLINDERS OF UF₆ PER YEAR
### DEPLETED URANIUM HEXAFLUORIDE MANAGEMENT PROGRAM

#### ASSAYS OF DEPLETED UF₆

<table>
<thead>
<tr>
<th>Assay Range</th>
<th>Metric Tons UF₆</th>
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</thead>
<tbody>
<tr>
<td>&lt; 0.30% U₂³⁵</td>
<td>272,000</td>
</tr>
<tr>
<td>0.30 - 0.41% U₂³⁵</td>
<td>260,000</td>
</tr>
<tr>
<td>0.41 - 0.71% U₂³⁵</td>
<td>28,000</td>
</tr>
</tbody>
</table>
DEPLETED URANIUM HEXAFLUORIDE
MANAGEMENT PROGRAM

PROGRAM STATUS

- WHERE WERE WE?
  * CYLINDERS ON GROUND, UNINSPECTED
  * CONVERSION IN 2020 (MAYBE)
  * OHIO SEEKING CONTROL UNDER RCRA
DEPLETED URANIUM HEXAFLUORIDE MANAGEMENT PROGRAM

PROGRAM STATUS
(CONTINUED)

- WHERE ARE WE?
  - SAFETY IMPROVEMENTS WELL UNDERWAY
  - PENDING APPROVAL OF OHIO SETTLEMENT
  - DEFENSE BOARD RECOMMENDATION DONE
DEPLETED URANIUM HEXAFLUORIDE MANAGEMNT PROGRAM

PROGRAM STATUS
(CONTINUED)

• WHERE ARE WE GOING
  * FINISH CYLINDER AND YARD IMPROVEMENTS
  * RESOLVE OHIO NOVs
  * COMPLETE PEIS AND IMPLEMENT ROD
### Depleted Uranium Hexafluoride Management Program

#### Current Cylinder Management
- Inspection
- Maintenance
- Surveillance
- Painting
- Valve Monitoring
- Yard Refurbishment
- Cylinder Relocation
- Development

#### Long-Term Planning
- Phase I: Strategy Selection - PEIS
- Phase II: Implementation - Record of Decision
DEPLETED URANIUM HEXAFLUORIDE MANAGEMENT PROGRAM

PHASE I
STRATEGY SELECTION

- TECHNOLOGY ASSESSMENT
- ENGINEERING ANALYSIS REPORT
- COST ANALYSIS REPORT
- PROGRAMMATIC EIS
- RECORD OF DECISION

PHASE II
IMPLEMENTATION

- 2nd TIER NEPA
- USES FOR DEPLETED URANIUM AND FLUORINE EVOLVE
- INDUSTRY RESPONSIBLE FOR PROCESSING, MANUFACTURE, & USE
DEPLETED URANIUM HEXAFLUORIDE MANAGEMENT PROGRAM

VERY PRELIMINARY COST ANALYSES

CONVERSION TO $U_3O_8$ - $2$ TO $6$ PER KG UF$_6$

CONVERSION TO METAL - $5$ TO $12$ PER KG UF$_6$

DISPOSAL AS OXIDE - $1$ TO $2$ PER KG UF$_6$

TRANSPORTATION - ABOUT $1$ PER KG UF$_6$

USE AS OXIDE OR METAL - $6$ TO $10$ PER KG UF$_6$

N.B. COST ANALYSIS REPORT PROVIDES ONLY COMPARATIVE DATA FOR STRATEGIES, NOT FOR BUDGETS OR BID ESTIMATES.
DEPLETED URANIUM HEXAFLUORIDE MANAGEMENT PROGRAM

PROGRAM ISSUES

• BECAUSE OF ITS SIZE, ANY PROCESSING OR TRANSPORTATION OF THE DEPLETED UF6 INVENTORY WILL BE VERY EXPENSIVE.

• NE CANNOT EXPECT SIGNIFICANT INCREASES IN BUDGETS.

• INDUSTRIAL INVESTMENT IS ESSENTIAL TO PROCESSING, MANUFACTURE, AND USE.
DEPLETED URANIUM HEXAFLUORIDE MANAGEMENT PROGRAM

CONCLUSIONS

- THE DEPLETED URANIUM AND THE FLUORINE ARE DEPARTMENT ASSETS.

- WE EXPECT THAT THERE WILL BE MORE THAN ONE USE.

- WE MUST WORK CLOSELY WITH INDUSTRY TO DEVELOP MUTUALLY BENEFICIAL ARRANGEMENTS.
EM OST Program
Introductory Remarks
and Program Overview

YMPO, NE and EM
Workshop on the Potential Uses of
Depleted Uranium in the DOE
Geologic Repository

Carl R. Cooley, EM 52
July 15 -- 17, 1997
Presentation Outline

• EM OST Role in the Depleted Uranium Program
  – Purpose
  – Sponsored Research*
  – Summary of Findings

• Conclusions

*Most EM OST work was conducted ahead of or in parallel to NE EIS Engineering Studies. Some results differ.

EM OST Efforts Began in 1993

• If DU were declared a waste, then EM could inherit a large problem
  – Significant inventories of oxide and metal at Fernald, Hanford and the Savannah River Site in addition to UF₆ at enrichment plants

• Major initial thrust was to determine if OST should sponsor research to reduce the cost of future management
EM OST Initial Efforts Were To Define the Size ($) of the Problem

- Estimate the cost of disposal
- Estimated the impact of RCRA
- Identify commercial uranium processing capabilities
- Evaluate potential shielding applications
- Identify cost reduction opportunities
- Perform market assessment to identify new applications or uses

Baseline Estimate of Conversion to Oxide, Packaging, Transportation and Disposal

$3.2 to $10.9 billion*

- Major impact on disposal costs was RCRA
  - Large incentive to avoid “waste” declaration
- Lower disposal cost assumed disposal at NTS in shallow land burial w/o stabilization
  - used available conversion cost data, many uncertainties
  - disposal location, costs, waste form, etc.
- Clearly a large liability for DOE
- Research on potential cost savings was justified

*T. J. Hertzler, et al, Depleted Uranium Disposal Options, EGG-MS-11297, May 1994
Technics Development Corporation
Performed the
Commercial Capabilities Survey*

- Concluded reuse of uranium metal for shielding was cost effective as an alternative to disposal
- Commercial US industry was fragile and further major reductions in capability were likely
- Chemical Conversion Costs -- UF₆ to Umetal - $8.80/kg U
- Fabrication Costs
  - Remelt, Metal Plate Forming - $4.40 to $8.80 per kg U
  - Milling, Machining, Welding - $11 to $22 per kg U

*Technics Development Corporation, A Summary of Trip Reports, Strategic Issues, and Commercial Site Data for the Uranium Recycle Planning Study, TDC-100, Rev 1, October 1993

SNL Study Evaluated Concept for DU Metal Casks at SRS for HLW Storage*

- Substituted casks for a planned second Glass Waste Storage Building at SRS
- DU metal casks used to store, ship, and dispose of HLW canisters
- Disposal of complete cask in repository assumed
- Evaluated casks with 1, 4 & 7 HLW canisters
- Cost effectiveness from reduced transportation of canisters in 7 canister cask and disposal credits

SAIC Conceptual Design Study Developed DU Cask for Storage, Transportation and Disposal*

- Design basis was the MPC program requirements
- Used DU metal sandwiched between inconel 825 and stainless steel (93 ton dry storage weight)
- Concept intended for repository disposal and provides corrosion allowance (2” of 304 SS, 3.5 “uranium, & 0.5 inch of Alloy 825)
- Prepared conceptual design cost of about $1.5 million w/o disposal credits**

*T. J. Hertzler, et al, Depleted Uranium Management Alternatives, EGG-MS-11416, August 1994
** Costs are probably under estimated as Castor cast iron casks now exceed $2 million

Opportunities for Cost Reduction Were Focused in Two Areas

- Los Alamos and INEL identified concepts for alternative, low cost uranium metal production methods
  - Large and lengthy development program needed to demonstrate potential viability
  - Even if successful, high DU metal fabrication costs still a deterrent to DU metal cask deployment
- An INEEL concept to incorporate depleted uranium aggregate into concrete
**INEEL Evaluated DU Metal Production Using Plasma Process***

- Concept used plasma heat source to directly reduce UF₆ in presence of hydrogen
- Continuous process, no waste streams (MgF₂)
- Produces U metal and anhydrous HF
- Successful bench scale demonstration
- Major cost reduction estimated at ~ 65% for large scale implementation
- Large program required to bring to pilot scale


---

**DU Metal Cask Estimates Were Costly for SNF Storage**

- Metal cost reduction potential identified but fabrication costs still very high
- Commercial interest in metal recycle from DU metal industry was high
- But, commercial storage cask vendors not interested (utilities will only pay for concrete)
- Spent fuel storage market is focused on low-cost concrete storage cask systems (SNC, VECTRA, Holtec, NAC)
DU Metal Casks Are Best Suited For Muti-Purpose Use

- Metal matrix makes combining storage and transportation technical feasible, but
- Since DU metal not an ASME code material, added steel needed to meet transportation requirements
- But, the high cost of a metal cask means higher up front costs and much higher Net Present Value to utilities
- Thus, utilities not willing to pay for technical benefits of metal casks

DUCRETE Concrete Was Conceived at INEEL as a Low Cost Way to Use DU in SNF or HLW Storage Casks

- INEEL initial physics work showed excellent shielding properties
- SNF storage cask vendors contracted to evaluate concept (Sierra Nuclear and Packaging Technology)
- Conducted limited proof-of-concept tests to demonstrate concrete properties using depleted uranium fuel pellets
- Challenge was to develop a low cost process for converting $\text{UF}_6$ to aggregate
EM OST Supported the Development of DUAGG at INEEL*

- Requirements:
  - DUAGG density should exceed 8 gm/cm$^3$
  - Chemically stable in oxidizing environment, stable in concrete matrix
  - Minimal leaching of DU from matrix
  - Mechanical properties suitable for use in concrete (DUCRETE Concrete)
  - Low cost fabrication method

*Program assumed UF$_6$ conversion to oxide was well developed industry capability

INEEL Developed Aggregate Production Process Which Met Requirements

- Aggregate density exceeds 8.5 gm/cm$^3$
- Stable in oxidizing environment to 150 C (further tests pending)
- Excellent leach results in ANS 16.1 tests
- Aggregate ceramic properties suitable for use in concrete
- Production costs estimated at $0.25 to $0.35 per lb for 80 ton per day plant capacity
Conversion of UF$_6$ to Oxide Form is Major Cost Element

- Conversion cost from UF$_6$ to Uoxide is major cost component -- $0.92 /\text{lb UO}_2$ (NE cost estimate)*
- Aggregate production is additional cost --
  - $0.30/\text{lb for aggregate production (80 ton/day plant, 80% Uoxide)},$
- 21 PWR storage cask
  - DUAGG cost ~ 46 tons -- $65K for UF$_6$ to Aggregate
- Total DUCRETE Cask Cost (Factory Fabrication) -- $214K$
  * 1991 Cogema estimate-$1.68/\text{lb for oxide (4.20/kg U)}$ used in EM studies

DUCRETE SNF Storage Casks
Appeared to Have Lowest Total Cost -- $214K$

- Based on low cost concrete technology
- Design similar to commercial concrete storage systems
- Excluding cost of UF$_6$ conversion to Oxide, cask cost is about $180K*$
- Large scale production estimate -- 9500 units

*1997 cost estimate from NMI-Chem Nuclear. EM 1994 cost estimate of ~ $500K
Engineering Work with DUCRETE Concrete
Limited to Compression Testing

- Compression test data done on original proof-of-concept samples (fuel pellets)
- Compression testing recently completed on semi-prototypic samples (hand pressed DUAGG)
- Additional engineering work on DUCRETE contingent upon production of large quantities of DUAGG from pilot scale facility (In Start Up)
- Latest compression test data typical of construction grade concrete (4000 to 5000 psi)

DU Poly Was Conceived After the DUCRETE Process as a Possible means to Incorporate Uranium Oxide Into Polyethylene

- Japanese have polymer based shielding products incorporating other heavy materials (Pb, Fe, etc)
- BNL successfully incorporated DU oxide into PE but product density not very high (~ 4 gm/cm3)
- Polyethylene stable to $10^8$ Rad
- Higher mass loading limited by product viscosity and high extrusion pressure
- Need to use DUAGG aggregate concept to increase loading fraction of DU oxide in PE matrix
Market Studies Did Not Identify Any New Uses for DU*

- HLW and SNF shielding was only credible and major use found
- Other ideas were either not viable or cost of DU metal made them impractical or quantities used were too small to impact DU inventory

* Reference, ORNL Kaplan Market Survey

Cost Reduction Opportunities Could Have Large Payback

- Continuous metalothermic reduction or plasma quench reduction technologies could lower cost of DU metal production
- Large scale DU metal casting could be integrated with metal continuous conversion to reduce DU metal fabrication costs
- Lower cost oxide production has already reduced the cost of DUCRETE concrete option compared to earlier EM studies
The UF₆ Represents a Large Federal Liability for Disposal

- EM estimated at $3 to $11 billion using earlier published data
- Current NE estimates $800K to $2.1 billion ($2.14 to $5.56 /kg U for disposal)

*Still Large by Any Measure!*

---

EM, NE and Industry Have Developed Independent Cost Estimates for Cask Systems and/or Disposal Costs

- **DU Metal System**
  - $649K¹ to $1.5 M²
  - 41.8¹ to 47.4² MTU
  - Disposal cost¹ - $100K to $260K
  - Cask net cost range with disposal credit - $550K to $1.25 M

- **DUCRETE System**
  - $546K¹ to $214K²
  - 39.5¹ to 37² MTU
  - Disposal cost¹ - $80K to $200K
  - Cask net cost range with disposal credit - $470K to ZERO

¹ NE Study; ² SAIC Study

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1- NE Study; 2- NMI Study
Conclusions
EM 50 Programs Have Provided Perspective on Potential DU Uses and Costs

- DU Metal storage casks are technically most versatile
  - But, at a significantly higher initial capital cost
- DUCRETE Concrete casks appear to provide the lowest SNF and HLW storage cost
- DOE liabilities for disposal are significant portion of total cask fabrication cost
- Final disposal location is needed
- Integration with repository design to bring synergistic benefits is the optimum solution
Overview of Mined Geological Disposal System (MGDS) Design and Operation

Presented to:
Workshop on Uses of Depleted Uranium in Storage, Transportation, and Repository Facilities
Las Vegas, Nevada

Presented by:
Paul Harrington, DOE-YMSCO
Viability Assessment Team Leader

July 15, 1997
Civilian Radioactive Waste Management System Concept

TYPES OF WASTE

GOVERNMENT-OWNED SPENT FUEL CUSTODIAN SITES

PURCHASER AND PRODUCER HIGH-LEVEL WASTE SITES

COMMERCIAL SPENT FUEL SITES
- Prepare Waste Description Information
- Load Canisters or Casks
- On-site Storage
- Prepare for Transport

Accept and Transport Waste

INTERIM STORAGE OPTION
- Transfer SNF Assemblies to Canisters
- Transfer Canisters to Storage Modules
- Store
- Retrieve and Prepare for Transport
- Close Interim Storage Facility

Transport

REPOSITORY
- Remove SNF Assemblies from Cask and Canisters (if necessary)
- Load and Seal Waste Packages
- Emplace Waste Packages
- Monitor and Confirm Performance
- Retrieve (if required)
- Close Repository
North Portal Area

Radiologically Controlled Area
- WHB Waste Handling Building
- WTB Waste Treatment Building
- CPB Carrier Preparation Building
- TMB Transporter Maintenance Building
- DCRS Disposal Container Receiving Shed

Balance of Plant Area
1. Warehouse
2. Shops
3. Administration
4. Mock-up
5. Utility
6. Fire/Medical
7. Service Station
8. Security Stations

Security Stations
Ground Support System

PRECAST CONCRETE LINING

CAST-IN-PLACE CONCRETE LINING

ROCK BOLT & MESH SUPPORT
Waste Packaging Operations

From Waste Receipt

Carrier Unloading

Canister Transfer

Assembly Transfer

Disposal Container Welding

Transporter Loading

To Emplacing Operations
Disposal Container Handling System

DC CELL

WASTE PACKAGE TRANSFER/DECON

WASTE PACKAGE TRANSPORTER LOAD
Waste Emplacement System

LEGEND

1. remote operations
2. emplacement drift entrance
3. waste package
4. final location
Engineered Barrier System Design Options

cast-in-place concrete or steel sets
drift lining

emplacement drift excavated surface

air gap (capillary barrier)

ceramic coating inside or outside waste package

backfill
drip shield supported by WP

waste package support assembly

invert with additives
Backfill Emplacement System

- mobile belt conveyor unit
- traveling gantry shuttlecar
- emplacement drift
- waste package (typ)
- temporary storage gantry shuttlecar
- stower
- backfill
- main haulage shuttlecar
- turnout
OUTER BARRIER LID
(A 516)

OUTER BARRIER
(A 516)

INNER BARRIER
(ALLOY 625)

INTERLOCKING PLATES
(STAINLESS STEEL BORON)

OUTER BARRIER LID
(A 516)

INNER BARRIER LID
(ALLOY 625)

SUPPORT GUIDES
(A 516)

LENGTH = 5335 mm
DIAMETER = 1604 mm
TARE WEIGHT = 31856 kg
LOADED WEIGHT = 46424 kg

44-BWR UCF
WASTE PACKAGE ASSEMBLY
5 DHLW/DOE SNF WASTE PACKAGE ASSEMBLY

LENGTH = 3790mm OR 5370mm
DIAMETER = 1970mm
TARE WEIGHT = 24,782kg
CANISTERED FUEL-WASTE PACKAGE ASSEMBLY

LENGTH = 5657 mm
DIAMETER = 1946 mm
TARE WEIGHT = 34,576 kg
LOADED WEIGHT = 68,351 kg
Facilitating DU Disposal

If EM wants to permanently dispose of the remaining DU, RW could facilitate the effort, but it would increase costs and could raise logistical, NEPA, RCRA, and licensing concerns.

[A 1993 M&O study on the utilization of DU in the CRWMS concluded that there is no compelling reason for using massive amounts of DU in the repository but there are strong arguments against such use. (Ref.: HQV.930924.0008) ]

DU disposed of within all waste packages or in drifts needs to be in a specified chemical and physical form to ensure proper long-term performance; waste packages may need to be larger and stronger to accommodate the DU; and the invert or backfill construction will be more complex and more costly.
Facilitating DU Disposal
(Continued)

- Logistical - DU availability would have to be synchronized with repository construction and emplacement schedules
  - Requires lag storage or coordinated deliveries
- Potential for EM EIS on DU disposition and for revision of the RW EIS for repository
- Requires review to determine if disposal of DU exceeds the scope of the NWPA
- Must satisfy applicable health requirements
Required Processing for Disposal

- $\text{UF}_6$ (volatile solid, BP= 56 C) must be processed into a more stable form (e.g., oxide, carbide, silicate) for any disposal scenario.

- $\text{UF}_4$ (a non-volatile solid) must be processed to remove fluoride for repository disposal.

- $\text{UO}_3$ (a stable powder) must be processed into a non-dusting form for repository disposal.

- DU Metal may need to be formed into shapes compatible with waste package designs.
Potential Applications for DU

- **Filler for Isotopic Dilution of HEU** -
  About 5,000 MTU of DU could be used in waste packages to isotopically dilute 150 MTU of HEU contained in DOE SNF. This would reduce the long-term criticality risk. The form of the DU would be metal, oxide, carbide, or silicate beads.

- **Shielding** -
  Uranium metal is an efficient shielding material, so perhaps 4,000 MTU could be used in transportation casks or a greater amount could be used as DUCRETE for storage casks.

- **Invert or Backfill** -
  Remaining 580,000 MTU of UF₆ could be converted and used in invert or backfill in the form of DUCRETE or silicate beads to further reduce criticality potential.
Engineering Considerations for the Use of DUCRETE Shielded Waste Packages for SNF and HLW

Prepared for the

DOE Workshop on the Uses of Depleted Uranium in the Repository

July 15-17, 1997

by

Bill Quapp, Paul DeLozier, & Martin Haas

7/12/97
This presentation provides the results of preliminary studies performed to establish:

- technical feasibility,
- nuclear performance,
- performance benefits and impacts,
- costs, etc.

of using DUCRETE shielded waste packages in the repository
DUCRETE Concrete is Simply a Conventional Concrete Using Heavy Uranium Ceramic Aggregate

- High density concrete (steel shot and iron ore aggregate) used in reactor construction and many other applications
- DUCRETE concrete uses an aggregate fabricated from depleted uranium oxide
- Resultant density is over 400 lb/ft³
- High Z and low Z material in DUCRETE Concrete provides excellent shielding characteristics

Comparison of Wall Thickness to Attenuate Neutron and Gamma Doses to 10 mR/h From 24 PWR Spent Fuel Assemblies

Dr. J. Sterbentz, INEEL
7/12/97
DUCRETE Concrete Achieves Gamma and Neutron Shielding in a Single Material Matrix

- Concrete is a low cost, high strength material
- Concrete properties can be tailored for strength
- Depleted uranium aggregate made with low cost processes
- DUCRETE concrete can be engineered to meet nuclear requirements

- Depleted uranium aggregate provides nuclear design flexibility
  - Can add neutron absorbers to aggregate (boron, gadolina, etc.)
  - Can adjust the aggregate density

Packaging Technology* Conducted Early Conceptual Design Studies

- Developed segmented fuel storage cask concept similar to MPC concrete overpack
- Natural convection cooling through air annulus

*Richard Haelsig
More Detailed DUCRETE Conceptual Design Studies Performed by Sierra Nuclear

- Developed DUCRETE version of SNC VSC-24
  - 24 PWR assemblies
  - 65 BWR assemblies
- Established dimensions, weights, external dose, etc.
- Identified sensitivity to DUAGG (depleted uranium aggregate) density
- Used Baseline SNC VSC-24 Licensing Methods
  - PWR fuel, 5 years cooling, 0.467 MTU / assembly, 35,000 MWD/MTU
  - PWR fuel with 55,000 MWD/MTU with 12.5 year cooling
  - 24 kW heat generation

SNC Conceptual Design Study Set Total Cask Weight Budget at 100 Tons

- At 100 ton wet weight, cask is capable of being loaded in the fuel pool for 88% of utilities
- Evaluated DUCRETE Cask shielding performance versus aggregate density and volume fraction
- Set wall thickness to keep wet weight at 100 ton
- Performed parametric sensitivity analysis of aggregate loading fraction, effect of sand vs. colmanite ore, effect of aggregate density, effect of total weight, and effect of inner steel wall thickness
- Used MCNP Monte-Carlo shielding code
Results from SNC Conceptual Design Study for 100 Ton Weight Weight Cask System

**Study Assumptions:**
- 24 PWR Assemblies
- 35 Gd/MU
- 5 year cooling time

![Graph showing dose rate vs. aggregate volume fraction with various markers for Silica Sand and Dolomite Sand.](image)

**SNC Conceptual Design**

**DUCRETE Dry Fuel Storage Cask**

- Right circular cylinder
- Steel shells (0.5 inch) on each side of DUCRETE concrete
- Shells provide strength and contamination control
- DUCRETE concrete adds to system strength
- Holds 24 PWR, 65 BWR Fuel Assemblies
- 89 inch OD
- 67 inch ID
- 62.5 inch basket OD
- 2 inch radial air gap
- 180 inch deep cavity
- 100 ton wet weight (fuel, basket, water)
- External dose 50 to 75 mR/h
The DUCRETE cask is 35 tons lighter and 40 in. smaller in diameter than casks made from ordinary concrete.

Primary Conclusions Support Feasibility of DUCRETE Concrete for Dry Fuel Storage Cask Applications

- Wide range of design options capable of meeting shielding requirements
- Based upon design studies, clearly able to meet requirements for other concepts
  - Smaller 21 PWR system
  - Inclusion of disposal overpack*

*Will not preserve 100 ton limit if sized to accommodate disposal package.
DUCRETE Cask Concept Allows Major Changes in Dry Fuel Storage Cask Manufacturing and Use

- Small diameter allows for rail transport
- Rail transport allows for factory fabrication
- Factory fabrication allows for utilities to “order casks as needed”
- Eliminates field construction of casks (Q/A problems)
- Allows transport from utility site to DOE Interim Storage Site and Geologic Repository (Allows reuse of storage system)

DUCRETE Overpack Cask Is Rail Transportable

Concrete Casks

DUCRETE Casks

DUCRETE is an Advanced Nuclear Shielding Material
The Quantity of Uranium Used in Casks Depends on Opening Date of Repository

If Yucca Mtn opens in 2010, 1300 storage containers required

DOE Inventory of DU sufficient for ~ 8700 containers

Plus - USEC is generating DU at 15,000 MT/y (375 containers/y)

Additional casks needed for HLW and DOE fuel

The Use of DUCRETE Casks For Reactor Dry Fuel Storage May Provide a Cost Effective Solution for Managing the DOE UF$_6$

- DOE’s inventory of depleted uranium is recycled rather than disposed (Avoids declaration as waste and higher management costs)

So, How Does This Relate to the Repository?
1. Cask Disposal Location Still Needed
2. Cask may provide advantages to repository design
Transportable DUCRETE Casks Can Be Used at Multiple Locations

- Utility Storage Site
- DOE Interim Storage Site, and
- Repository as Lag Storage, and
- To Create a Shielded Waste Package

Other Drivers for the Repository Design

The NWTRB* suggested...

"At least one alternative should be based on conservative application of existing technology, for example, shielding the waste packages so that human activity near them is possible and ventilating the emplacement tunnels to provide temperatures low enough for effective functioning of humans and machines, sensors, and other equipment."

*1996 NWTRB Report to Congress
Largest Benefit of DUCRETE Casks May Be As a Self Shielded Waste Package

- Addresses the NWTRB suggestion for a contact handled option
- Provides a UF₆ management solution
- Can provide cost savings to total spent fuel and UF₆ management challenge
- May reduce repository system capital cost and simplify long term operations
- Reduced near term technical uncertainty may accelerate licensing process and reduce litigation

Many DUCRETE Cask Design Options Are Possible

Option 1
- **Reactor Storage, Interim Storage and Disposal Overpack**
  - Heaviest cask
  - Very low external radiation dose (<1 mR/h)
  - Lowest lifecycle cost

Option 2
- **Repository Only Overpack**
  - Lightest weight
  - Optimized dose design
  - Lowest initial cost
  - Does not offer benefits to Reactor Storage or Interim Storage Site

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DUCRETE Multipurpose Cask

Characteristics

(Option 1)

- Features
  - 75 in ID, 97 in OD
  - 180 in cavity length
  - 86 tons empty weight
  - 30.5 ton disposal package
  - Total weight 137 ton

- Requires large quantity of
  - DU
  - 61 tons of DUAGG
  - 80 tons of UF6 equivalent
  - DUCRETE density of 6.32 gm/cm³.

With both disposal package and DUCRETE Overpack, the external radiation will be <1 mR/h.
DUCRETE Cask Can Be Designed For Use Only As Repository Overpack (Option 2)

- Features
  - 75 in ID, 97 in OD
  - 180 in cavity length
  - 42 tons empty weight
  - 30.5 ton disposal package
  - Total weight 93 ton

- Requires smaller quantity of DU
  - 27 tons of DUAGG
  - 35 tons of UF6 equivalent
  - DUCRETE density of 5.15 gm/cm³.

*The external radiation dose for the most limiting fuel loading will be ~25 mR/h*

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External Contact Dose from DUCRETE Shielded Waste Package

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[Graph showing dose rate versus DUCRETE thickness]

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Overpack Design Can Be Optimize to Meet Dose Goals

![Graph showing the relationship between mass and DUCRETE thickness.](image)

Estimated Cask Weight, Dose, and Fabrication Cost Are A Function of the Specific Application

![Bar chart showing loaded weight, cost, and external dose for different applications.](image)
At least Three Variations for Overpack Emplacement Are Envisioned

- Vertical in the drift
  - minimum change in overpack, requires modification of the drift
- Vertical in a mega-drift
  - minimum change in overpack, major change in repository design
- Horizontal in drift
  - requires change in overpack heat rejection method

Self-Shielded Disposal Overpacks
Using DUCRETE Concrete Overpacks in Drifts

[Diagram of concrete overpack with dimensions: 216 in. height, 200 in. diameter, 162 in. diameter for the drift]
Self-Shielded Disposal Overpacks Using DUCRETE Concrete Overpacks in "Super-Drift"

Self-Shielded Disposal Overpacks Using DUCRETE Concrete Overpacks in Drifts
The DUCRETE Overpack Benefits
Design and Operation of Repository

- Reduces near term materials corrosion and prediction issues
- Provides extremely low radiation dose external to cask
- Facilitates waste package inspection
- Eliminates remote operations & improves personnel safety
- Provides drip shield for long term performance
- Mitigates criticality concerns (large inventory of U238 for dilution of fissile material and neutron absorbers can be added to aggregate if desired)
- May reduce total system cost based on net present value considerations

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Major Repository Impacts

- Changes repository design basis
- May require modification of drift emplacement concept
- Requires larger ventilation system to maintain temperature less than ~ 104 F (40C)
- Requires more robust emplacement equipment
- Requires an operating crew after 40 year emplacement phase for another 60 years (?)
DUCRETE Overpack Concept is Responsive to NWTRB Recommendation for a Low Technology Alternative

- Elimination of remote operations simplifies emplacement activities
- Simplifies predictability of near term operations
- Facilitates fuel canister emplacement and retrieval
- Simplifies environment for human and sensor operation
- Removes water from repository

Eliminating One or Two Years of Delay Might Pay for Long Term Operations

- For a annual program cost of $300 million, a one year delay is equivalent to the Net Present Value of an annual operating budget of $9.5 million per year for 100 years
- Consequently, design simplifications (such as ventilation, self shielded disposal packages) might expedite licensing and reduce litigation
- Expedited permitting might pay for the extended operating cost
Conclusions

• DUCRETE Shielded Waste Package can be used to simplify repository design and operations
• Simpler operating environment may result in easier licensing and will provide greater personnel safety
• Avoidance of licensing or litigation delay may offset the expenses for long term operation
• Many questions remain unanswered but --
  -- the purpose of this workshop is to begin the discussion
DEPLETED URANIUM OXIDE FILL FOR FUEL PACKAGES

Allen G. Croff
Oak Ridge National Laboratory

Workshop on Uses of Depleted Uranium in Storage, Transportation, and Repository Facilities

July 15, 1998
Las Vegas, Nevada

Prepared by the
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managed by
LOCKHEED MARTIETTA ENERGY RESEARCH CORP.
for the
U.S. DEPARTMENT OF ENERGY
under contract DE-AC05-84OR21400
Outline

- Concept description
- Fill characteristics
- Benefits
- Outstanding issues
- Variants and elaborations
Baseline Concept Description

- Load spent fuel assemblies into a MPC
- Inject DU fill into void space with vibration to achieve maximum density
- Seal MPC
- Manage according to standard plan
Fill Characteristics

- DUO₂ for compatibility with cladding and maximum benefits
- A mix of small (<1mm) particles to maximize density
- ~3 Mg DU/Mg spent fuel heavy metal
  - ~32 Mg DU or ~37Mg DUO₂ per 21-assembly package
Benefits

- Shielding: Weight-effective reduction of package surface dose (primarily gamma) by \(~10X\)
- Package Criticality: Significantly reduces probability of water-moderated criticality
- Structural: Provides internal support to prevent package collapse
- Post-Closure Leachate Access: Fill oxidizes and swells first, inhibiting water access to fuel
- Post-Closure Fuel Dissolution
  - Fill oxidation helps to maintain beneficial reducing conditions
  - Entering groundwater is saturated with uranium, inhibiting fuel dissolution
Benefits (cont’d)

- Post-Closure Criticality: Isotopically dilutes uranium plus decayed plutonium to non-critical enrichments
Outstanding Issues

- Fill characteristics
  - Optimal mix of particle sizes
  - Fill gas
- Fill emplacement techniques
  - Have full-scale Canadian experience
  - Effects of vibration
  - Open MPC versus small opening in lid
- Bulk thermal conductivity of fill and thermal impacts
  - Very preliminary calculations indicate temperatures are acceptable
Outstanding Issues (cont’d)

- Incorporation into repository system baseline and system engineering
- Impacts on transportation
  - Net weight change
  - Effects on transportation package performance and certification
- Post-closure performance impacts
  - Supportable credit for benefits of an engineered barrier
  - Radiological impacts of DU
- Cost/benefit
Variants and Elaborations

- Use of fills other than UO$_2$
  - Uranium silicate glass beads have been studied (ORNL/TM-13045)
- Modification to enhance neutron shielding
- Use of U$_x$O$_y$ for backfill or invert
DUAGG and DUCRETE Properties

Workshop on the Potential Uses of Depleted Uranium
Presented by: Robert E. Heard
July 15 -- 17, 1997
What is DUAGG and DUCRETE

- DUAGG - Depleted Uranium Oxide aggregate. Uranium oxide power (UO$_2$, U$_3$O$_8$, and/or UO$_3$) that has been mixed with artificial basalt precursors, pressed, then liquid phase sintered into a dense body aggregate

- DUCRETE - Depleted Uranium Oxide concrete. Construction grade concrete containing DUAGG as a replacement for the natural rock aggregate
Background Information

- Proof of principle development began in FY-94
- Initial development was performed by hand grinding and mixing of material
- Test results for this early development showed aggregate could be favorably produced
  - Bulk density 8.59g/cm³
  - Open porosity of 2.78%
- DUCRETE samples produced from UO₂ fuel pellets and crushed sintered U₃O₈ chips
  - Compressive strength tests - 15% - 20% less strength than the reference conventional concrete samples
Background Information (cont’d)

- FY-95 development included UO₂ oxide as the starting powder
- Hand milling was still used to grind and mix the material
- Sintering in a reducing atmosphere maintained UO₂ as the primary phase
- Leach testing showed no detectable leaching of Uranium
- Boron was introduced as an additive to improve neutron shielding characteristics
  - Bulk density decreased (limit established at 8.0 g/cm³ or greater)
  - Open porosity less than 2%
Background Information (cont’d)

- FY-96 development concentrated on aggregate composition optimization
- FY-97 development concentrating on aggregate production
  - Additional testing to establish characteristics of aggregate
    - DUAGG oxidation testing
    - DUCRETE strength testing
    - DUCRETE shielding tests
  - Using automated process for producing DUAGG
Chemical Composition

- DUAGG composition is 80 volume percent $\text{UO}_2$ and 20 volume percent additives
- Additives are:
  - Soil
  - Clay
  - Mill additions
  - Boron*

* Note: Other shielding material could be used
Neutron/Gamma Shielding Properties

- Modeling efforts based on cask design assumptions
  - Cask weight limit of 100 tons
  - 8.0g/cm³ aggregate density
  - 0.5 inch inner and outer steel liners
  - Cask payload of 24 moderate burnup (35GWD), 5 year cooled PWR fuel assemblies
  - 45% DUAGG volume fraction
- Peak surface dose rate calculated to be between 50 and 100 mrem/hr
- Actual tests to be conducted this summer
Leach Characteristics

- Leach testing conducted in FY-95 on material produced at that time
- Inductively Coupled Plasma (ICP) instrument was used to analyze the leachate solutions
- This instrument has a lower detection limit of 0.5 ppm
- Most DUAGG samples showed uranium ion levels below the detection limit
- Exception - 2 samples showed uranium ion levels from 0.5363 to 9.2320 ppm
Thermal Properties

- Thermal properties have not been analyzed but are assumed to be similar to ordinary concrete casks
- DUCRETE materials testing for cask designing will include thermal properties testing
  - Thermal emissivity
  - Thermal conductivity
  - Heat capacity
  - Coefficient of thermal expansion
Mechanical (Strength) Properties

- Tested in compression strength according to ASTM C 39-72, "Compressive Strength of Cylindrical Concrete Specimens"
- Tested at 100°C and 150°C, test results show
  - no degrading of strength
  - only two samples of DUCRETE were tested at each temperature
- Additional testing to be conducted at 250°C later this year
Fabrication Experience

- Fabrication process for DUAGG has been established by the INEEL
  - Calcination and Milling
  - Drying and Low Temperature Calcination
  - Binder Addition/Agglomeration
  - Pressing/Briquetting
  - Drying
  - Sintering
- Hand pressing performed at INEEL with Briquetting performed by subcontractor
- Subcontract to Nuclear Metals, Inc. to produce approximately 6 tons of aggregate
Schedule Requirements

- DUCRETE materials testing to support cast design
  - One to two years to complete
  - Dependent on supply of aggregate
- DUCRETE cask development
  - Four years plus to have a licensed and compliant cask
APPENDIX F

Potential Depleted Uranium Uses in the DOE Geologic Repository

1. DUCRETE Casks for At-Reactor Storage and Disposal of DUCRETE Casks (w/o Fuel) in Empty Buffer Drifts

2. DU Metal Casks for At-Reactor Storage and Disposal of DU Metal Casks (w/o Fuel) in Empty Drifts

3. DUCRETE Casks for Dry Fuel Storage and Self-Shielded Disposal Packages

4. DU Metal Casks for Dry Fuel Storage and Self-Shielded Disposal Packages

5. DUCRETE Inverts in Repository Drifts

6. Oxide Aggregate (DUAGG) as Backfill Around Disposal Package

7. DU Oxide Powder as Backfill Around Disposal Package

8. DU Oxide or DUSCOBS as Filler Inside of Fuel Canister

9. DUCRETE Shield Blocks Positioned Next to Unshielded Disposal Package

10. DUCRETE Half-Cylindrical Shield with End Ventilation

11. DUCRETE Cask Integrated with Disposal Package

12. DU Metal Cask Integrated with Disposal Package

13. Disposal of U₃O₈ Directly into Repository Empty Buffer Drifts

14. Disposal of UO₂ DUAGG Directly into Repository Empty Buffer Drifts

15. Keyhole Design with Shielding Cover

16. DUCRETE Concrete Drift Liner

17. Use of Depleted Uranium in Shield Doors

18. Use of Depleted Uranium in High-level Waste/High Enriched Uranium Disposal Packages
APPENDIX G

Pre-Closure Working Group Report

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Martin Haas, Yucca Mountain M&O (MK)    Jim Dubrin, LLNL
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Richard Yoshimura, SNL                     Charlotte Johnson, SAIC (scribe)
Mark Fortsch, TRW

Workshop participants were divided into two groups for focused discussions associated with pre-closure and post-closure requirements of the mined geologic repository and potential uses of depleted uranium in storage, transportation and repository facilities. The discussions were aimed at identifying the advantages and disadvantages of potential beneficial uses of depleted uranium for repository applications.

The pre-closure and post-closure periods are defined as follows. Pre-closure is the operations phase of the geologic repository, including any retrieval of materials emplaced in the repository. Post-closure begins when the repository is permanently closed and there is no further intervention by man (except for monitoring). The current projected schedule for the repository is shown below.

2005 - 2010: Construction
2010 - 2033: Waste Emplacement, Operations (ongoing construction, also)
2033 - 2110: Caretaker Phase
2110: Beginning of Decommissioning
2110 - 2125: Closure (when no one works there any more = post-closure)

Therefore, the pre-closure period would include 2005 through 2125 and post-closure would begin at the end of the closure activities (e.g., after 2125).

Eighteen options (plus several variations of select options) for the use of depleted uranium were discussed in the pre-closure work group. A brief description of each option follows, along with a discussion about the potential benefits, drawbacks, obstacles and uncertainties, regulatory impacts, and suggestions for overcoming the obstacles.

Option 1. DUCRETE Casks for At-Reactor Storage and Disposal of DUCRETE Casks (w/o Fuel) in Empty Buffer Drifts

This option uses the DOE depleted uranium hexafluoride by first converting it to an oxide that is then manufactured into a ceramic aggregate. The aggregate is subsequently used in DUCRETE concrete dry fuel storage casks. DUCRETE concrete is made by replacing the conventional aggregate in concrete with a heavy aggregate made from depleted uranium oxide. The DUCRETE concrete has an estimated
density of over 400 lb/ft³. The depleted uranium in the high density concrete provides gamma ray attenuation and the hydrogen in the chemically bound water provides neutron attenuation. The DUCRETE concrete can be used in place of ordinary concrete for storage casks. The cask concept is similar to conventional concrete casks produced by dry fuel storage cask vendors such as Sierra Nuclear, NAC, and Holtek. The higher density, DUCRETE casks are smaller diameter and weigh less than casks made from conventional concrete. These attributes make transportability possible. Conventional concrete casks are manufactured on the utility site and used with no provision for transport elsewhere because they exceed the width requirements for unrestricted rail transport. Comparative diameters of a conventional and DUCRETE concrete cask are about 135 inches and 90 to 100 inches, respectively. DUCRETE casks are also estimated to weigh about 35 tons less than conventional concrete casks. The weight of an empty cask is about 65 tons, which is within the load capacity of rail systems.

The DUCRETE casks can be used at the utility sites for interim storage, shipped from the utility to the DOE interim storage facility when it becomes available and finally shipped to the geological repository when it is available to receive fuel (for lag storage). The DUCRETE casks are not used to ship fuel but only to store it at the various interim storage sites. If not needed for lag storage at the repository, it can be re-deployed at a utility site or at a DOE interim storage facility. When no longer needed for fuel storage, the casks may be disposed of in the repository by using any remaining buffer drifts and excavating additional drift space, as required. The casks could also be disposed during the closure period by using the area available in the entrance and exit drifts. As there is essentially no heat load from the depleted uranium in the DUCRETE cask, these can be packed as closely as physically possible in the repository.

This activity is included as a pre-closure activity since construction of drifts/tunnels, etc. would occur before closure.

A significant number of casks (estimated 1000 - 2000 casks) would be available for emplacement in empty buffer drifts (based on a 2010 opening of the repository). There would be only 5 drifts available for use during the “caretaker” period (during closure). Casks could probably not be placed between disposal waste packages (e.g., those already planned for disposal) since larger drifts would be required to be constructed than is currently planned. It should be noted that the 5 drifts will have a purpose in the repository until closure even if they are not used for cask emplacement.

The development of a DUCRETE cask would have to be accelerated to be able to take advantage of its use in the spent nuclear fuel management program. If they are not ready for use for ten more years, the window of opportunity for use in the spent fuel management program will have been missed.

The purpose for placement is disposal of depleted uranium casks that had been used elsewhere (i.e., no benefits to repository). The repository does not have operational requirements for the use of DUCRETE casks.

Benefits
• DUCRETE casks provide a disposition solution for depleted uranium (enough inventory for about 9000 casks total, only about 1/5 of these would be used if the repository opens according the schedule). Disposal in the repository provides a solution for final disposition of the casks.

Drawbacks

• Casks would need space in the repository.
• Casks would require storage on surface at repository.
• There would be an operational cost of emplacement.
• There would be no operational advantages to the repository (i.e., no shielding advantages).

Obstacles/uncertainties associated with implementing

1. DUCRETE cask must be developed to the point it is ready to be used.
2. Total number of casks is not known.
3. Is retrieve ability required?
Regulatory issues

Although the repository could conceivably offer a cost effective disposal alternative for DUCRETE casks, the current law [Nuclear Waste Policy Act [NWPA]] prohibits disposal of any material other than spent nuclear fuel and high-level waste. Without a change in the law, the disposal of casks would probably be in violation of the NWPA, since the empty DUCRETE casks would be classed as low level waste.

The NRC has stated that oxide is the preferred form for disposal of depleted uranium in a repository. Therefore, there would be no regulatory concern about disposal form.

The Yucca Mountain license application may have to be modified (heavy metal limits related to spent nuclear fuel disposal), although initial analysis does not support this need.

A performance assessment (PA) long-term source impact (long-term impact to the public, including groundwater) is needed.

What needs to be done to address/resolve obstacles/uncertainties

1. Need performance assessment, source term study
2. Need cost impact study on repository
3. Need a deployment decision for the depleted uranium (use in the form of casks)

Option 2. DU Metal Casks for At-Reactor Storage and Disposal of DU Metal Casks (w/o Fuel) in Empty Drifts

This option is essentially similar to option 1, above. The differences are highlighted here. Depleted uranium (DU) metal has been used for shielded casks for many years. This concept uses DU metal sandwiched between stainless steel shells. The stainless steel provides the required strength since DU metal, while strong, is not an American Society for Mechanical Engineers (ASME) code material. It requires no development effort, although DUCRETE does require development. DU metal has been used in this type of application for licensed transportation casks for many years. However, the cost to produce DU metal is many times higher than estimated for DUCRETE concrete. Consequently, casks would be more expensive to fabricate than currently estimated for DUCRETE concrete. It is expected that the same types of cost differences between DUCRETE and DU metal would exist as between conventional concrete casks and metal ones. An advantage of DU metal casks over DUCRETE is that the thermal conductivity of the material would preclude the need for natural circulation venting of the cask. The DU metal casks would be lighter and smaller than DUCRETE concrete casks, however; this feature may be of little value as both are rail transportable.

Disposal of DU metal casks does not meet previously identified Nuclear Regulatory Commission (NRC) requirements for disposal as an oxide. However, this disposal mode should be environmentally acceptable. The only known issue may be associated with the future potential corrosion of the depleted
uranium upon contact with water and the resultant formation of hydrogen gas. This issue could probably be mitigated by repository design.

This activity is included as a pre-closure activity since construction of drifts/tunnels, etc. would occur before closure.

Purpose for placement is disposal of DU metal casks that had been used elsewhere (i.e., no benefits to repository). The repository does not have operational requirements for the use of DU metal casks.

1000 - 2000 casks would be available for emplacement in empty buffer drifts. There would be only 5 drifts available for use during the “caretaker” period (during closure). Casks could probably not be placed between disposal casks since larger drifts would be required to be constructed.

Benefits

- Use of DU metal casks provides a disposition solution for depleted uranium (enough inventory for about 9000 casks total, only about 1/5 of these would be used depending upon the opening schedule of the repository). Disposal in the repository provides a solution for final disposition of the casks.

Drawbacks

- Casks would need space in Yucca Mountain.
- Casks would require storage on surface at the repository.
- There would be an operational cost of emplacement.
- There would be no operational advantages to repository (e.g., no shielding advantages).
Obstacles/uncertainties associated with implementing

1. Total number of casks is not known.
2. Is retrieve ability required?
3. Additional obstacles/uncertainties (to those included in option 1, above) are those associated with disposal of uranium metal.

Regulatory issues

Although the repository could conceivably offer a cost effective disposal alternative for DUCRETE casks, the current law (Nuclear Waste Policy Act [NWPA]) prohibits disposal of any material other than spent nuclear fuel and high-level waste. Without a change in the law, the disposal of casks would probably be in violation of the NWPA, since the empty DUCRETE casks would be classed as low level waste.

NRC has stated that oxide is the preferred form for disposal in a repository. Therefore, there could be a regulatory concern about the metal disposal form. However, there should not be a technical limitation for the use of metal.

The Yucca Mountain license application may have to be modified (heavy metal limits related to spent nuclear fuel disposal), although initial analysis does not support this need.

Still need performance assessment (PA) long-term source impact (long-term impact to the public, including groundwater).

What needs to be done to address/resolve obstacles/uncertainties

1. Evaluate processes to reduce the cost of the DU metal system.
2. Need performance assessment, source term study
3. Need cost impact study on repository
4. Need a deployment decision for the depleted uranium (use in the form of casks)
5. Look at Nuclear Waste Policy Act related to heavy metal impact.

Option 3. DUCRETE Casks for Dry Fuel Storage (At-Reactor or CIF) and Self-Shielded Disposal Packages

This option is similar to option 1, above, except that the final disposition would be in the drift space as part of a shielded waste package. This option brings each DUCRETE cask to the repository. The fuel canister is fitted with the corrosion resistant disposal package and reinserted into the DUCRETE cask (the cask would be oversized when fabricated for eventual insertion of the disposal package at the repository). The cask and its contents are then moved into position in a suitable location underground using a horizontal rail car system. Since the packages would be fully shielded, no remote operations are required. Due to the overall package size, some drift modifications may be necessary or desirable.
depending upon the emplacement method (rail versus gantry). Alternative concepts for disposal of the shielded package are also possible, such as mining of a large "room" for disposal in tightly packed groups. If horizontal disposal in drifts were selected, modifications to the natural convection cooling of the casks would be required during cask design (current storage casks use vented cooling designs for cask vertical orientation).

The over pack weighs 86 tons empty and 137 tons loaded. The fuel canister and waste package go inside, so the package is essentially an over pack. This is a disposal option starting life at an at-reactor location or interim storage facility. Package is designed oversized to later accept the waste package (Must have full shielding at the spent fuel interim storage facility, and therefore it is over shielded when the package is combined with the waste package). The fuel canister would be removed from the transportation cask, placed in the disposal package, and that system placed in the over pack for disposal as a unit. DUCRETE fully shields neutrons and gamma radiation.

Benefits

- Remote handling equipment is not required for emplacement in the repository (this extends removability over the life of the operation of the facility since are still able to move around).
- Concept requires ventilation to reduce ambient air temperature. Ambient air temperature provides other benefits including reduced corrosion due to reduced moisture and reduced temperature, and more predictability (not stressing the Mountain). There are no performance studies of extended operation of a mine at an elevated temperature (so lowering the temperature to a more typical mine operational temperature may have some positive benefits).
- Retrieval is simplified due to contact handling.
- Periodic inspection is facilitated by shielding and drift cooling.
- Gamma and neutron shielding is provided.

Drawbacks

- More ventilation drifts would be required (see obstacles, below); therefore, cost is increased.
- Larger drifts may have to be drilled (due to size of package).
- Heavier and larger packages would be handled; therefore, heavier handling equipment may be required.
- Visual inspection of waste packages might be difficult. However, the access issues are probably offset by the remote inspection requirement of the high radiation and high temperature baseline inspection requirement.
- The cask would have to be sized appropriately relatively early and before repository waste package size is finalized.

Obstacles/uncertainties associated with implementing

1. Thermal evaluation of the package is needed (air going into emplacement drift is about 27°C, and exiting is about 50°C), including a look at the capability of the package to handle this temperature
differential. A heat transfer study is needed to identify thermal response of horizontally vented casks.

2. The rate at which drifts are currently scheduled to be ventilated is 280 cubic meters/second. Approximately 2000 cubic meters/second would be required. A lot more main drifts would have to be driven and more vertical shafts would be required to the surface. The capital cost is estimated at about $600,000,000 for increased ventilation requirements (tunnels). Ventilation is forced with fans. A more complete assessment is needed to optimize cooling requirements with higher air humidity and to obtain better cost estimates.

3. There may be some DUCRETE performance issues for extended use in the repository.

4. The over pack design must be integrated with the waste package design.

Regulatory issues

1. Moving from a passive ventilation system to a forced system will possibly create some additional regulatory hurdles (e.g., licensing).
2. Simplifying retrieval operations may have some regulatory benefits.

What needs to be done to address/resolve obstacles/uncertainties

1. Must prove can make a viable over pack within allotted time period.
2. Need detailed ventilation study.
3. Need detailed over pack heat transfer studies.
4. May need emplacement system redesign.
5. Over pack design must be integrated with waste package design.

Option 3A.  DUCRETE Casks for Self-Shielded Disposal Packages (use in repository only)

This option is identical to option 3, above, except that the over pack is designed to provide shielding with the waste package installed. The fuel canister is fitted with the corrosion resistant disposal package and reinserted into the DUCRETE cask. The cask and its contents are then moved into position in a suitable location underground using a horizontal rail car system. Since the packages would be fully shielded, no remote operations are required. Due to the overall package size, some drift modifications may be necessary or desirable depending upon the emplacement method (rail versus gantry). Alternative concepts for disposal of the shielded package are also possible, such as mining of a large “room” for disposal in tightly packed groups. If horizontal disposal in drifts were selected, modifications to the natural convection cooling of the casks would be required during cask design (current storage casks use vented cooling designs for cask vertical orientation).

The over pack weighs 42 tons empty and 93 tons loaded. The fuel canister goes inside, so the package is essentially an over pack. Over pack and waste package all go into disposal facility as a unit. DUCRETE fully shields neutron and gamma radiation.

Benefits
• Remote handling equipment is not required for emplacement in the repository (this extends removability over the life of the operation of the facility since are still able to move around).
• Concept requires ventilation to reduce ambient air temperature. Ambient air temperature provides other benefits including reduced corrosion due to reduced moisture and reduced temperature, and more predictability (not stressing the Mountain). There are no performance studies of extended operation of a mine at an elevated temperature (so lowering the temperature to a more typical mine operational temperature may have some positive benefits).
• Retrieval is simplified due to contact handling.
• Gamma and neutron shielding is provided.
• Periodic inspection is facilitated by shielding and drift cooling.

Drawbacks

• More ventilation drifts would be required for ventilation system (see obstacles, below); therefore, cost is increased.
• Larger drifts may have to be drilled (due to size of package).
• Heavier and larger packages would be handled; therefore, heavier handling equipment may be required.
• Visual inspection of waste packages might be difficult. However, the access issues are probably offset by the remote inspection requirement of the high radiation and high temperature baseline inspection requirement.
• The cask would have to be sized appropriately relatively early and before repository waste package size is finalized.

Obstacles/uncertainties associated with implementing

1. Thermal evaluation of the package is needed (air going into emplacement drift is about 27 C, and exiting is about 50°C), including a look at the capability of the package to handle this temperature differential. A heat transfer study is needed to identify thermal response of horizontally vented casks.
2. The rate at which drifts are currently scheduled to be ventilated is 280 cubic meters/second. Approximately 2000 cubic meters/second would be required. A lot more main drifts would have to be driven and more vertical shafts would be required to the surface. The capital cost is estimated to be about $600 million for increased ventilation requirements (tunnels). Ventilation is forced with fans. A more complete assessment is needed to optimize cooling requirements with higher air humidity.
3. There may be some DUCRETE performance issues for extended use in the repository.
4. The over pack design must be integrated with the waste package design.

Regulatory issues

Moving from a passive ventilation system to a forced system will possibly create some additional regulatory hurdles (e.g., licensing).
Obstacles/uncertainties associated with implementing

1. Thermal evaluation of the package is needed (air going into emplacement drift is about 27°C, and exiting is about 50°C), including a look at the capability of the package to handle this temperature differential. A heat transfer study is needed to identify thermal response of horizontally vented casks.

2. The rate at which drifts are currently scheduled to be ventilated is 280 cubic meters/second. Approximately 2000 cubic meters/second would be required. A lot more main drifts would have to be driven and more vertical shafts would be required to the surface. The capital cost is estimated to be about $600 million for increased ventilation requirements (tunnels).

Ventilation is forced with fans. A more complete assessment is needed to optimize cooling requirements with higher air humidity.

3. There may be some DUCRETE performance issues for extended use in the repository.

4. The over pack design must be integrated with the waste package design.

Regulatory issues

Moving from a passive ventilation system to a forced system will possibly create some additional regulatory hurdles (e.g., licensing).

What needs to be done to address/resolve obstacles/uncertainties

1. Must prove can make a viable container within allotted time period.

2. Need detailed ventilation study.

3. Need detailed heat transfer studies.

4. May need emplacement system redesign.

5. Over pack design must be integrated with waste package design.

Option 4. DU Metal Casks for Dry Fuel Storage and Self-Shielded Disposal Packages

The DU metal over pack for spent nuclear fuel or high-level waste is loaded at the reactor, fuel consolidation site or the high-level waste facility. It would be stored as long as necessary. When the repository is ready to accept fuel or high-level waste, the shielded over pack would then be placed in a transport over pack, and would be shipped to the repository. At the repository, the shielded over pack would be removed, still containing the spend nuclear fuel or high-level waste, loaded into a disposal package, and placed in the repository.

This option is similar to option 2, above, except the over pack is disposed with the waste package. This is also similar to option 3, but the over pack is made using DU metal rather than DUCRETE concrete. The fuel canister is fitted with the corrosion resistant disposal package and either 1) reinserted into the DU metal cask (the cask would be oversized when fabricated for eventual insertion of the disposal package at the repository) or 2) the entire DU metal cask would be inserted into the disposal waste package. The cask and its contents are then moved into position in a suitable location underground using a horizontal rail system. Since the packages are fully shielded, no remote
operations are required. Due to the overall package size, some drift modifications may be necessary or desirable. Alternative concepts for disposal of the shielded package are also possible, such as mining of a large “room” for disposal in tightly package groups.

Neutron shielding would normally be provided by polyethylene during the reactor or CISF use. Since repository regulations prohibit combustible materials, the polyethylene might have to be removed and consequently, the DU metal cask may not provide adequate neutron shielding.

The cost to convert UF₆ to a fabricated metal end product is estimated at about $20.00 per kilogram.

Benefits

- DU metal will make a lighter over pack than DUCRETE concrete. However, if the waste package is located around the DU metal over pack, it will be heavier--probably offsetting any weight advantage of the DU metal cask. If the waste package is placed inside of the DU metal over pack, the cost will increase significantly because of the DU needed for the oversized over pack.
- Relative to option 3, above, the heat transfer system would be marginally better.

Drawbacks

- The waste package diameter would have to be enlarged to accommodate shielded cask.
- This package will not provide adequate neutron shielding.
- Other drawbacks would include those listed for option 2 and 3, above.

Obstacles/uncertainties associated with implementing

1. Ventilation requirements still would need to be decided.
2. The adequacy of the heat transfer system would have to be evaluated.

Regulatory issues

See 2 and 3, above.

What needs to be done to address/resolve obstacles/uncertainties

1. Evaluate criticality issue: might be a better reflector; therefore, increasing short term criticality potential.
2. Need heat transfer study
4. Need cost evaluation.
5. Evaluate alternate/additional neutron shielding materials.
6. Evaluate processes to reduce the cost of the DU metal system.
7. Alternate neutron shielding material needs to be identified.
8. Evaluation methods for reducing DU metal costs need to be identified.
Option 4A. DU Metal Casks for Self-Shielded Disposal Packages

DU metal over pack for spent nuclear fuel or high-level waste is loaded at the reactor, fuel consolidation site or the high-level waste facility. It is stored as long as necessary. When the repository is ready to accept fuel or high-level waste, the shielded over pack is then placed in a transport over pack, and shipped to the repository. At the repository, the shielded over pack is removed, still containing the spent nuclear fuel or high-level waste, loaded into a disposal package, and placed in the repository.

This option is also similar to option 2 and 3, above. Since the packages are fully shielded, no remote operations are required. Due to the overall package size, some drift modifications may be necessary or desirable. Alternative concepts for disposal of the shielded package are also possible, such as mining of a large “room” for disposal in tightly package groups.

This concept is applicable to all DOE high-level waste, spent nuclear fuel and commercial fuels. There is an approximate match between DU availability and needs. Waste packages could be outside (waste package could be placed around depleted uranium). Neutron shielding may not be as good. Fully shielded disposal packages may not meet personnel exposure limits as explained in option 4.

Cost estimated to convert UF₆ to fabricated metal end product at about $20.00 per kilogram.

Benefits

- The benefits for option 4A are the same as those for option 4, plus the following additional advantages. The canister is loaded at the repository site, the fuel or high-level waste would remain in it, therefore greatly reducing the handling cost.
- Relative to option 3, above, this package may have a marginally better heat transfer system.

Drawbacks

- The waste package diameter would have to be enlarged to accommodate shielded cask.
- The waste package will not provide adequate neutron shielding.
- Other drawbacks include those listed for option 2, 3 and 4, above.

Obstacles/uncertainties associated with implementing

1. Ventilation requirements: to be decided.
2. Adequacy of heat transfer system would have to be evaluated.

Regulatory issues

See option 2, above.

What needs to be done to address/resolve obstacles/uncertainties
1. Evaluate criticality issue: might be a better reflector; therefore, increasing short term criticality potential.
2. Need heat transfer study
4. Need cost evaluation.
5. Evaluate alternate/additional neutron shielding materials.
6. Evaluate processes to reduce the cost of the DU metal system.

Option 5. **DUCRETE Inverts in Repository Drifts**

This option uses the depleted uranium aggregate developed for DUCRETE concrete and substitutes for the conventional aggregate in the manufacturing of drift inverts.

**Benefits**
- There would be no benefits for the pre-closure period.

**Drawbacks**
- Radiation inhalation dose may be some (but this is not a big issue).
- Deployment will require some health and safety precautions to avoid personnel exposure from inhalation.
- Cost is a factor (DUCRETE could be replaced with plain concrete). Would need to determine if benefits would outweigh increased cost.

**Obstacles/uncertainties associated with implementing**

Since no pre-closure benefits were identified, obstacles and uncertainties are not discussed.

**Regulatory issues**

Since no pre-closure benefits were identified, regulatory issues are not discussed.

**What needs to be done to address/resolve obstacles/uncertainties**

Since no benefits were identified, resolution of obstacles/uncertainties are not discussed.

Option 6. **DUAGG as Backfill Around Disposal Package**

The option uses the depleted uranium aggregate (DUAGG) developed for DUCRETE concrete and deposits it around the disposal package for dilution of the fissile uranium in the future. This option could use all of the DOE depleted uranium.

**Benefits**

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• There would be no benefits to the repository during the pre-closure period.

**Drawbacks**

• Potential radiation inhalation dose may be more than in option 5, above.
• Concern is included in deployment aspect of backfill (health physics/industrial hygiene) -- heavy metal included in backfill -- critical organ sensitivity (kidney).

**Obstacles/uncertainties associated with implementing**

Since no pre-closure benefits were identified, obstacles and uncertainties are not discussed.

**Regulatory issues**

Since no pre-closure benefits were identified, regulatory issues are not discussed.

**What needs to be done to address/resolve obstacles/uncertainties**

Since no pre-closure benefits were identified, resolution of obstacles/uncertainties are not discussed.

**Option 7.  DU Oxide as Backfill Around Disposal Package**

This option is similar to option 6, above, except the uranium oxide is deposited directly around the disposal package for dilution of the fissile uranium in the distant future. This option could use all of the DOE depleted uranium.

**Benefits**

• No benefits were identified for use during the pre-closure period.

**Drawbacks**

• Potential radiation inhalation dose may be higher than option 6, above.
Obstacles/uncertainties associated with implementing

1. Concern is included in deployment aspect of backfill (health physics/industrial hygiene) -- heavy metal included in backfill -- critical organ sensitivity (kidney).
2. Since no benefits were identified, obstacles and uncertainties are not discussed further.

Regulatory issues

Since no pre-closure benefits were identified, regulatory issues are not discussed.

What needs to be done to address/resolve obstacles/uncertainties

Since no pre-closure benefits were identified, resolution of obstacles/uncertainties are not discussed.

Option 8. DU Oxide or DUSCOBS as Filler Inside of Fuel Canister

This option uses depleted uranium in an oxide or small glass particle geometry. The material is poured into the canister at the repository. Conceptually, the purpose is to provide enhanced self-shielding of the fuel canister and to provide material in intimate contact with the fuel to reduce risks of future criticality during postulated flooding events and in the post disposal package failure time when the fuel elements are predicted to be dissolved by groundwater. It is postulated that the presence of this oxide would delay the dissolution of the fuel elements because the groundwater would become saturated from the backfill oxide until it was totally dissolved and removed from the waste package.

Benefits

- No pre-closure benefits.
- Potential post-closure benefit: Collapse of waste package may be delayed or prevented. Further analysis is required.
- Post-closure benefit would include criticality control.

Drawbacks

- Heat transfer problem is likely in the pre-closure period (cladding corrosion problem potential).
- May produce unacceptable fuel cladding mechanical stresses.
- Even distribution of material will be difficult.
- Complexity and cost of mechanical vibration to add particulate material.
- Would require surface facility modifications for remote filling of the particulate.
- Would add to complexity of remote emplacement operation due to large weight addition in the disposal package.
Obstacles/uncertainties associated with implementing

1. VERY early stages of development in the United States.
2. Use of DU oxide of DUSCOBS as filler inside of the fuel canister would add to the complexity of remote operations within the repository.

Regulatory issues

Additional analysis and development required for regulatory approval.

What needs to be done to address/resolve obstacles/uncertainties

Lots of development required to resolve issues associated with the pre-closure period.

Option 9. DUCRETE Shield Blocks Positioned Next to Unshielded Disposal Package

This option uses DUCRETE concrete to fabricate shield blocks for use in the repository. One postulated use would be to surround the emplaced disposal packages. If suitably surrounded, future operations and maintenance might be simplified. Alternatively, shield blocks (in a box-like geometry) could accompany the disposal canister during emplacement and provide for partial shielding. Ventilation requirements would preclude complete encasement. DUCRETE concrete shield blocks would be about 1/3 the thickness of conventional concrete shield blocks. Blocks here could include small walls and other useful shapes.

Benefits

- Shielding is improved compared to standard waste package, potential dose reduction (probably by a factor of about 10) — some scattering going on; about 10 R/hr at surface of container, about 1R/hr for individual near the waste package (bottom line is there would not be enough self-shielding).
- **Minimal pre- and no post-closure benefits identified; therefore, no further evaluation necessary.**
- Could have short term entrances to repair packages (for very short time period)

Drawbacks

- Since minimal pre- and no post-closure benefits were identified, no further analysis of drawbacks was required.
Obstacles/uncertainties associated with implementing

Since minimal pre- and no post-closure benefits were identified, obstacles/uncertainties are not discussed.

Regulatory issues

Since minimal pre- and no post-closure benefits were identified, regulatory issues are not discussed.

What needs to be done to address/resolve obstacles/uncertainties

Since minimal pre- and no post-closure benefits were identified, resolution of obstacles/uncertainties is not discussed.
Option 10.  **DUCRETE Half-Cylindrical Shield with End Ventilation**

This option is a variation of option 9, above. It uses a DUCRETE half cylinder (cylinder sliced axially through the center line) for shielding emplaced disposal packages. The concept came to be called a “carport” during the discussions at the workshop. The concept involves casting a depression in the invert blocks for accommodating some portion of the cylinder. Pre-cast DUCRETE concrete blocks might provide a portion of the disposal package sidewall height. Finally, a half-cylinder would be emplaced covering the remainder of the disposal package. The length of the shield would extend beyond the waste package a distance to minimize direct dose to personnel in the drift. This distance needs to be determined. Ventilation air would be required to remove heat through the ends. There would be no DUCRETE shielding on the end of this structure.

**Benefits**

- The concept provides a drip shield.
- Additional rock fall protection is provided.
- The package could be designed to be completely self-shielded.
- Inspection (although indirect) might be easier than in the cask because more space on the ends would be available for remote inspection equipment.
- It may be possible to move packages over one another because the drip shield would protect the packages.
- It would be possible to consider placing the “carport” on separate rail (so it would not have to be lifted).
- Ventilation may be improved in the shaft (compared to closed containers — carport type design is open and creates less obstacles to air flow).

**Drawbacks**

- There will be a need for additional drift ventilation.
- If a package had to be removed, all shields and packages would have to be moved out that are stored after it in the tunnel. More complex handling equipment will be required.

**Obstacles/uncertainties associated with implementing**

1. Will drip shield survive the length of time it would be required to survive?
2. Heat transfer during backfill and operational (caretaker) phase
3. Acceptability of backfilling against packages. Is it acceptable not to backfill the “carports?” Backfill is intended to protect packages when tunnels finally collapse. Does the “carport” prevent meeting the backfill requirements.
4. Can carport be designed that, if dropped, it would remain intact?
5. Operational impacts for potential retrieval must be evaluated.
Regulatory issues

Similar to other DUCRETE options.
Backfill question (see obstacles, above).

What needs to be done to address/resolve obstacles/uncertainties

1. Need thermal analysis of fuel cladding temperatures in “carport” and definition of required drift air flow.
2. Analyze effectiveness of radiation shield, especially scattering analysis.

Option 11. DUCRETE Cask Integrated with Disposal Package

This option would integrate the disposal package into the design of the dry fuel storage cask. In a normal concrete storage cask, up to 1.75 inches (45 mm) of steel lining is used to provide structural strength to the cask and to diffuse the heat load away from the concrete to prevent localized overheating. If the disposal package were to replace this cask component, total system weight and size reductions might be possible. Total system shielding requirements can be met at lower total weight than for option 3, above. The cask is an over pack that includes the disposal package for the corrosion protection. The current waste package would be redesigned to incorporate DUCRETE shielding into the package. Ventilation cooling of the drifts would be required.

INCLUDES ALL ISSUES AS 3, ABOVE, with the following exceptions/additions.

- Different cost detail.
- Separate inner liner for casks would not be required.
- Larger uncertainties.

Benefits

• Same as option 3, above.
• The total weight of the waste disposal package is lower.
• Less cask material is consumed and wasted.
• There is a higher weight for the on-site storage cask.

Drawbacks

• Same as option 3, above.
• Higher early cost for the on-site storage system.

Obstacles/uncertainties associated with implementing
1. See option 3, above.

**Regulatory issues**

See option 3, above.

**What needs to be done to address/resolve obstacles/uncertainties**

1. See option 3, above.

**Option 12. DU Metal Cask Integrated with Disposal Package**

This option would integrate the disposal package into the design of the DU metal dry fuel storage cask. The current waste package would be redesigned to incorporate depleted uranium metal into package for shielding (but does not change protection thickness). The disposal package would provide the interior steel shell. The DU metal would be sized to provide the added shielding. An external steel shell would provide the needed structural surface and attachment points for handling as well as a corrosion barrier for the depleted uranium metal. Neutron shielding at the reactor storage site would be provided by a polyethylene layer on the outside of the over pack. This layer may have to be removed before disposal due to limitations of combustible material in the repository. Neutron shielding effectiveness is uncertain without some hydrogenous material on the exterior of the over pack.

INCLUDES ALL ISSUES AS 4, ABOVE, with the following exceptions/additions.

- Different cost detail.
- Inner liner for casks would not be required.
- Larger uncertainties.

**Benefits**

- Same as for option 4, above.

**Drawbacks**

- Same as for option 4, above.

**Obstacles/uncertainties associated with implementing**

1. See option 4, above.

**Regulatory issues**

See option 4, above.
What needs to be done to address/resolve obstacles/uncertainties

1. See option 4, above.

Option 13. Disposal of \( \text{U}_3\text{O}_8 \) directly into Repository Empty Buffer Drifts

This option, similar to option 1, above, disposes of depleted uranium oxide directly in buffer drifts. The uranium oxide would be packaged in drums or boxes to facilitate handling. Such containers would have a limited lifetime but could be designed to last for the duration of the emplacement period. Containment during this period would be needed to contain the radon dose from the large quantity of depleted uranium and to eliminate risks from inhalation of depleted uranium. For packaging efficiency, some drift modifications might be necessary to maximize the volume of packaged uranium oxide. As there is essentially no heat load from the depleted uranium, the containers could be packed as closely as physically possible in the existing excavated drifts.

The disposal volume needed is about 150,000 cubic meters for 560,000 metric tons of depleted uranium. Compared to option 1 where empty casks are disposed, this option is easier to effect. Deployment is easier — B-25 boxes and 55-gallon drums are easier to handle than 65 ton casks. Must be containerized anyway to be shipped to Yucca Mountain. Equipment required to handle will be standard, e.g., forklifts. The oxide is assumed to be compacted (to about 3 g/cc). Volume is much greater than for option 14, below. Need to do drift scale analysis to determine impacts. PA analysis required — probably will assume packages are breached on day one (i.e., don’t take into account credit for the containers — want containers sufficient enough to carry through “caretaker” period because personnel exposures are not desired). Probably will need approximately eight to nine drifts. Could just stay ahead of the arriving inventory (by having drifts ready).

Benefits

• Similar to option 1, above.

Drawbacks

• No clear benefit to the repository.

Obstacles/uncertainties associated with implementing

Since no benefit was identified to the repository obstacles/uncertainties are not discussed.

Regulatory issues

Precluded by the Nuclear Waste Policy Act legislation limiting the repository to spent nuclear fuel and high-level waste.

What needs to be done to address/resolve obstacles/uncertainties
Since no benefit was identified to the repository, resolution of obstacles/uncertainties is not discussed.

Option 14. Disposal of UO₂, DUAGG Directly into Repository Empty Buffer Drifts

This option is similar to option 13, above, and differs only in the physical form of the depleted uranium disposed. This option disposes of depleted uranium oxide in an aggregate form -- DUAGG aggregate -- directly into buffer or specially mined drifts. The uranium oxide would be packaged in large heavy bags to facilitate handling. The DUAGG aggregate (density near 8 g/cc) would have a reduced volume compared to the unstabilized oxide. Packaged and compressed bulk oxide would have a density of about 2.8 to 3.0 g/cc. DUAGG aggregate would have a particle density of about 8.5 g/cc and, assuming a 60% packing density, the bulk density could be about 5.1 g/cc. Improved packaging efficiency might be obtained with a second size aggregate.

Considering that no containers are required other than large bags and the density advantage of DUAGG aggregate, essentially twice as much uranium could be disposed per unit volume of repository compared to bulk uranium oxide powder. The ceramic aggregate itself is expected to minimize radon release; however, data does not currently exist to support this. For packaging efficiency, some drift modifications might be necessary, but these are expected to be less than for bulk containerized oxide. As there is essentially no heat load from the depleted uranium, the bulk bags can be package as closely as physically possible in the existing excavated drifts.

Smaller volume than 13, above, about 75,000 cubic meters.

Benefits

• No clear benefits to the repository have been identified.

Drawbacks

• No clear benefits to the repository have been identified.

Obstacles/uncertainties associated with implementing

Since no benefit was identified to the repository obstacles/uncertainties are not discussed.

Regulatory issues

Since no benefit was identified to the repository, regulatory issues are not discussed.

What needs to be done to address/resolve obstacles/uncertainties

Since no benefit was identified to the repository, resolution of obstacles/uncertainties is not discussed.

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Option 15. Keyhole design with shielding cover

Benefits

• See option 10, above.
• Retrieval is easier.

Drawbacks

• Tunnel construction costs would be more expensive.
• Ventilation would be required.
• Heat transfer could be an issue.
• Moisture control could be an issue.

Obstacles/uncertainties associated with implementing

1. See option 10, above.

Regulatory issues

See option 10, above.

What needs to be done to address/resolve obstacles/uncertainties

See option 10, above.

Option 16. DUCRETE concrete drift liner

Benefits

• No pre-closure benefit to the repository has been identified.

Drawbacks

• No pre-closure benefit to the repository has been identified.
• See 5, above.

Obstacles/uncertainties associated with implementing

Since no benefit was identified to the repository obstacles/uncertainties are not discussed.

Regulatory issues
Since no benefit was identified to the repository, regulatory issues are not discussed.

**What needs to be done to address/resolve obstacles/uncertainties**

Since no benefit was identified to the repository, resolution of obstacles/uncertainties is not discussed.

**Option 17. Use of Depleted Uranium in Shield Doors**

The use of depleted uranium in shield doors for each of the 200 emplacement drifts was proposed as a possible use of depleted uranium. This could be in the form of either DU metal or as DUCRETE. Currently, the design includes the use of simple shadow shields at the end of each drift adjacent to the lifting pads. The shadow shields block direct radiation from the last waste package to the main drift. Each shadow shield consists of 24 inch thick barriers made of ordinary concrete that are slightly higher and wider than the last emplaced waste package. This in turn results in a maximum radiation field in the main drift of 0.80 mrem/hr. The average radiation field is expected to be less than 0.4 mrem/hr. The doors to each emplacement drift now consist of 1-inch thick steel containment barriers with vent controls included for ventilation control of air.

The current design permits considerable flexibility in emplacement and retrieval of waste packages with minimal risk. A major consideration in the current approach was to avoid the use of massive doors with the attendant problems associated with support and complex operation.

**Benefits**

- The use of DU in emplacement drift doors could reduce the maximum radiation field from 3.53 mrem/hr to essentially zero.
- No net benefit to the repository has been identified.
**Drawbacks**

- Massive frames would be required to support each door. Additionally, problems in opening such doors are anticipated in the event that they are not opened regularly due to settling problems.
- A major cost disadvantage is envisioned in that installed DU shield doors may cost approximately $500,000 each. The cost of the shadow shields is estimated to cost less than $10,000, a 50:1 penalty.
- Finally, the proposed use would not consume a significant amount of the total DU inventory.

**Obstacles/uncertainties associated with implementing**

Since no benefit was identified to the repository obstacles/uncertainties are not discussed.

**Regulatory issues**

Since no benefit was identified to the repository, regulatory issues are not discussed.

**What needs to be done to address/resolve obstacles/uncertainties**

Since no benefit was identified to the repository, resolution of obstacles/uncertainties is not discussed.

**Option 18. Use of Depleted Uranium in High-Level Waste/High Enriched Uranium disposal packages**

**Benefits**

- Moderator exclusion

**Drawbacks**

- DU would not do a good job by itself.
- Use in disposal packages would not use a lot of the depleted uranium inventory.

**Obstacles/uncertainties associated with implementing**

Since minimal benefit was identified to the repository obstacles/uncertainties are not discussed.

**Regulatory issues**

Since minimal benefit was identified to the repository, regulatory issues are not discussed.

**What needs to be done to address/resolve obstacles/uncertainties**

G-25
Since minimal benefit was identified to the repository, resolution of obstacles/uncertainties is not discussed.

CONCLUSION

Eighteen options, plus several variations to individual options, were evaluated. The options evaluated ranged in complexity and covered backfill options, invert options, shielding with ventilation, and over pack options. Those options that were identified as having benefits to the repository in the pre-closure period are listed below. These options may warrant further study related to beneficial use in the repository during the pre-closure period.

1. Over pack options (Options 3, 4, 11, and 12)
2. The DUCRETE half-cylindrical shield with end ventilation (also known as the “carport”), Option 10
APPENDIX H

Post-Closure Working Group Report

Group Chair -- Dave Stahl, TRW
Members -- Carl Cooley, DOE EM  Abe Van Luik, YMSCO  Ched Bradley, DOE-NE
Bob Heard, INEEL  Dave Haught, YMSCO  C. Brown, DOE-NE
E. Bernard, SNL  Hugh Benton, TRW  Steve Baker,
consultant  Gary Knight (scribe)  Allen Croff, ORNL
M. Rafferty, DOE-

Ports

Option 1 and 2.  DUCRETE and DU metal casks for storage and disposal and disposal of empty casks in repository

Benefits

DUCRETE and DU metal casks for storage and disposal and disposal of empty casks in repository. You could use them for isotopic dilution. Could take place of other nasty isotopes in mountain, or taking a “fast path” to replacing bad isotopes. Could put them in the end of each emplacement drift for shielding. Once all emplacement drifts are filled you could fill up access drifts with them.

Drawbacks

You could leach out some area under the repository so you could dilute some problems from fuel stored above it. No benefit by putting into repository spent storage casks. Gets rid of surface disposal facility for contaminated casks.

Obstacles/uncertainties associated with implementing

If it’s not HLW or SNF, is it permitted to go into the mountain?

Regulatory issues

Does DOE have go through a Part 61 regulatory finding (calling DU a low-level waste)?

What needs to be done to address/resolve obstacles/uncertainties

Questions need to be answered.

Options 3 and 4.  DUCRETE or DU metal casks for dry fuel storage

Benefits
DUCRETE or DU metal casks for dry fuel storage

**Drawbacks**

Uses up all DU inventory; but no benefit from post-closure viewpoint. Drawbacks: if you increased weight, rail system and gantry would have to be heavier and more sturdy. More overall waste packages would have to be required, because it will hold smaller number of fuel rods. Might not be able to remove packages not on the end, because of weight of casks. Upside -- criticality and chemical buffering. Preclosure benefit is the neutron shielding of DU casks. It would oxidize and mobilize faster than the SNF.

**Obstacles/uncertainties associated with implementing**

Thermal conductivity is an obstacle. Chemistry long-term is a big uncertainty. U238 to U235 ratio could be 600 to 1.

**Regulatory issues**

NRC license.

What needs to be done to address/resolve obstacles/uncertainties

Thermal testing of DUAGG and DUCRETE. Test for chemical effects.

**Option 5. DUAGG and DUCRETE inverts in repository drifts**

**Benefits**

DUAGG and DUCRETE inverts in repository drifts

**Drawbacks**

Criticality (slight) and chemical buffering to delay migration. Material could be used between the rails (in lieu of gravel) for increased criticality control.

**Obstacles/uncertainties associated with implementing**

Inverts are required anyway; this will provide invert material with benefits mentioned above. It would clearly be useful for a small number of waste packages, but it could be used throughout the entire repository but the benefits are reduced.

**Regulatory issues**

No known regulatory issues.
What needs to be done to address/resolve obstacles/uncertainties

Thermal conductivity.

**Options 6 and 7. DUAGG/DU oxide as backfill**

For any criticality issue, it is better to have the DU in the invert than in a backfill.

**Benefits**

DUAGG/DU oxide as backfill

**Drawbacks**

For possible benefit for criticality (slight) and chemical buffering to delay migration; but basic (.8 T/foot of DU/ft. of tunnel and have 100 miles of emplacement drifts) question is why in backfills at all, but in inverts? Using it in inverts is an early 21st century issue, using it in backfills is an early 22nd century issue.

**Obstacles/uncertainties associated with implementing**

No clear, established known benefits; lower priority; cost is against crushed tuff which is there in great quantity.

**Regulatory issues**

No outstanding regulatory issues.

What needs to be done to address/resolve obstacles/uncertainties

Testing to clarify chemical buffering benefit and to improve performance; need to test thermal conductivity of DUAGG and DUCRETE.

**Option 8. DU Oxide or DUSCOBS as Filler Inside of Fuel Canister**

Filling inside of fuel canister, going into the repository, to fill voids

**Benefits**

For DU, it is better to put it in commercial SNF to replace the water presently there, meaning less moderation and less likely for the uranium to swell limiting water flow and possibly leading to the fuel being damaged. Another benefit of putting it in canister is you do not have to deal with it handlingwise. Physical compatibility with fuel assemblies.
Drawbacks

Presently steel shot is better, as it is a better mnemonic reflector. Could have thermal problems. In terms of benefits, it is difficult to justify using it in large amounts to meet the standards required of the repository. Biggest postclosure concern is extra weight per waste package of ~ 15 tons/per.

Obstacles/uncertainties associated with implementing

Thermal, chemical, and mechanical uncertainties.

Regulatory issues

No regulatory issues.

What needs to be done to address/resolve obstacles/uncertainties

Need to explore thermal conductivity question.

Option 9. DUCRETE Shield Blocks next to Unshielded Disposal Package (pre-closure issue)

Option 10. DUCRETE Half-Cylindrical Shield with End Ventilation

Benefits

Protects waste package but an anti-corrosion sheath (i.e., ceramic) is needed to mitigate corrosion.

Drawbacks

Requires more and sturdier handling equipment to put shields over waste packages in drifts.

Obstacles/uncertainties associated with implementing

Chemistry and thermal conductivity.

Regulatory issues

None identified.

What needs to be done to address/resolve obstacles/uncertainties

Testing on chemistry and thermal issues.
Option 11 and 12. DUCRETE and DU Metal Cask Integrated with Disposal Package

**Benefits**

Uses up all DU inventory; but no benefit from post-closure viewpoint. Does provide criticality and chemical buffering. Preclosure benefit is the neutron shielding of DU casks. It would oxidize and mobilize faster than the SNF.

**Drawbacks**

If you increased weight (however, weight is less than options 3 and 4), rail system and gantry would have to be heavier and more sturdy. More overall waste packages would have to be required, because it will hold smaller number of fuel rods. Might not be able to remove packages not on the end, because of weight of casks.

**Obstacles/uncertainties associated with implementing**

Thermal conductivity is an obstacle. Chemistry long-term is a big uncertainty. U238 to U235 ratio could be 600 to 1.

**Regulatory issues**

NRC license required.

**What needs to be done to address/resolve obstacles/uncertainties**

Thermal testing of DUAGG and DUCRETE, although not as critical as options 3 and 4. Test for chemical effects.

Option 13. Disposal of U$_3$O$_8$ into Repository Empty Drifts (pre-closure issue)

Option 14. Disposal of UO$_2$ DUAGG into Repository Empty Drifts (pre-closure issue)

Option 15. Keyhole Concept

**Benefits**

Same benefits as the drip shield, plus it permits leapfrogging of waste packages desired to be retrieved.

**Drawbacks**

Have less material than inverts. Two boring machines are required to make it. Possibility that during post-closure package may be overheated.
Obstacles/uncertainties associated with implementing

You have restricted air flow and restricted space and the resultant impact on thermal issues. To solve both issues, you could install fans in each drift, but that is a great deal of fans.

Regulatory issues

NRC regulations.

What needs to be done to address/resolve obstacles/uncertainties

Thermal issues need study.

Option 16. Shielded Doors(pre-closure issue)

Option 17. DUCRETE Entombment

Benefits

Criticality and chemical buffering. Preclosure benefit is the neutron shielding of DUCRETE.

Drawbacks

If you increased weight, rail system and gantry would have to be heavier and more sturdy.

Obstacles/uncertainties associated with implementing

Thermal conductivity is an obstacle. Chemistry long-term is a big uncertainty. U-238 to U-235 ratio could be 600 to 1.

Regulatory Issues

NRC license.

What needs to be done to address/resolve obstacles/uncertainties

Thermal testing of DUAGG and DUCRETE. Test for chemical effects.

CONCLUSION

Eighteen options, plus several variations to individual options, were evaluated. The options evaluated ranged in complexity and covered backfill options, invert options, shielding with ventilation, and cask
options. Those options that were identified as having benefits to the repository in the post-closure period are listed below. These options may warrant further study related to beneficial use in the repository during the post-closure period.

1. Option 5, DU Inverts under the waste canisters
2. Options 6 and 7, Backfill around the canisters
3. Option 8, DU filler inside the canisters

Based on the benefits and consideration of the relative effectiveness of the options in achieving the desired benefits, the use of DU in the inverts supporting the waste packages appears to offer the greatest promise. The backfill and container filler options would also place DU in relatively close proximity to the SNF and it may be premature to eliminate these options from the preferred path. The DU would be added during prefabrication of the inverts, thus, simplifying the industrial hygiene and radiological controls issues associated with handling DU. Encapsulation of the DU in the inverts would help to bring the release of the DU to the repository environment into better congruence with the release of fissile uranium from the SNF. It would also be in the direction of gravity flow of water and it would not negatively impact the thermal performance of the waste packages or the repository.
APPENDIX I

NRC Letter
Docket No:  70-3070

Louisiana Energy Services, L.P.
ATTN:  W. Howard Arnold
       President
       2121 K Street, N.W.
       Suite 850
       Washington, DC  20037

Gentlemen:

Since disposition of depleted uranium (DU) tails is an important decommissioning licensing issue for the proposed Claiborne Enrichment Center, the Nuclear Regulatory Commission performed an assessment of the issues involved. Our evaluation assumes that the bulk of DU tails will eventually be disposed of as a waste. We examined the acceptability of disposal of the LES enrichment plant tails, as depleted UF₆, in a licensed 10 CFR Part 61 disposal facility as suggested by LES’s "Depleted Uranium Hexafluoride Management Study." We have completed our review of this proposal. Based on our analysis, we have reached the following conclusions.

The preferred chemical form for final disposition of the DU tails is U₃O₈, regardless of U-235 concentration. Even if stored tails were later further processed and depleted of U-235, the bulk of DU tails must still be disposed of. Compared with UF₆, U₃O₈ is the more stable physicochemical form and the more compatible, as regards to safety, with long-term disposition of tails. Conversion of the DUF₆ to DUF₄ for final disposition is not acceptable because its physicochemical, long-term stability is incompatible with final disposal under 10 CFR Part 61.

The Environmental Impact Statement (EIS) supporting 10 CFR Part 61 did not contemplate large volumes of DU tails. Our analysis, using methodology similar to that used for the Part 61 EIS, concludes that near-surface disposal of such large quantities of DU tails is not appropriate, both because of its potential radiological impact and its chemical toxicity. However, other disposal alternatives under 10 CFR Part 61 may be viable; e.g., deep mine disposal. Therefore, disposal options, other than near-surface disposal, must be considered for the DU tails. Disposal options must be accompanied with supporting analyses. The analyses should include funding provisions for storage, tails conversion to the oxide form, final disposition and, if applicable, transportation costs.

Your analyses should also consider an appropriate schedule for conversion and disposal. Since you are proposing to start production in phases, which may take several years, the conversion of DUF₆ to DUF₄, or other suitable waste form, should start 10 to 15 years after initiating production, or after generating 80,000 tons of tails, whichever is reached first.
In summary, demonstration of viable means of DU tails ultimate disposition and provision for financial assurance are needed. It is recognized that the total volume of waste to be generated for the LES Claiborne Enrichment Center is part of a much larger national inventory. Therefore, LES DU tails disposition may be addressed as part of the national inventory disposal scheme.

We would be pleased to discuss these matters further with you after you have considered them. If you have any questions, please contact Dr. Lidia A. Roche' at (301) 504-2695.

Sincerely,

Jerry J. Swift for
/S/

John W.N. Hickey, Chief
Fuel Cycle Safety Branch
Division of Industrial and Medical Safety
Office of Nuclear Material Safety and Safeguards

cc: Attached list
APPENDIX J

Additional Depleted Uranium Studies


2. “Uranium, ATSDR Public Health Statement”, December 1990, DOE-RW (This document was not retrieved for this report.)

3. “A Summary of DOE EM Efforts to Explore Recycle and Reuse of the Department’s Depleted Uranium Resource,” by Carl Cooley, DOE-EM

4. “Status Report: DUAGG/DUCRETE Experimental Program” by LMITCO

5. “DUAGG/DUCRETE Development Requirements” by LMITCO and NMI


8. “Background Information for DOE Workshop on the Potential Uses of DU in the Repository” by W. Quapp, P. DeLozier, and M. Haas

   a. DUCRETE Concrete Cask Description and Use
   b. DU Hexafluoride: The Source Material for Advanced Shielding Systems
   c. SRS High Level Waste Canister Storage System Using DUCRETE Overpacks
   d. Options for Using or Disposing of DU in the DOE Geologic Repository
The United States Department of Energy is currently engaged in a cooperative program with the Electric Power Research Institute (EPRI) to develop a spent nuclear fuel dry transfer system (DTS). The system will enable the transfer of individual spent nuclear fuel assemblies between a conventional top loading cask and multi-purpose canister in a shielded overpack, or accommodate spent nuclear fuel transfers between two conventional casks. Regulatory and prototype demonstration activities and results will be discussed. The DTS has several significant applications and could benefit the Department of Energy and commercial nuclear facility operators in a number of ways. It has the potential to:

- permit shutdown nuclear reactor sites to decommission pools;
- provide capability at interim storage facilities to transfer assemblies from small transportation casks to sealed canisters;
- provide capability at reactor sites with limited crane capacity to transfer assemblies into large packages to facilitate on-site storage in larger capacity casks;
- allow recovery operations at shutdown reactor sites with independent spent nuclear fuel storage installations;
- provide a means for utilities that can presently handle only a truck cask to utilize a rail cask;
- allow transfers of spent nuclear fuel from existing utility dual purpose systems into alternative systems if required, without returning to the reactor storage pool; and
- support existing and future Department of Energy spent nuclear fuel management activities.

The Electric Power Research Institute contracted with Transnuclear, Inc., Hawthorne, New York, to design the dry transfer system and prepare a topical safety analysis report. The Dry Transfer System for Spent Fuel: Project Report was completed in December 1995 and delivered to the Department of Energy. The Dry Transfer System Topical Safety Analysis Report was completed and delivered in August 1996. The Department of Energy submitted the topical safety analysis report to the United States Nuclear Regulatory Commission in September 1996 for review and approval. Approval of the topical report by the Commission is expected by year-end 1998 could expedite future licensing and use of the DTS.
A $4.5 million project was initiated in June 1996 to demonstrate a prototype of the DTS at the Department of Energy, Idaho National Engineering and Environmental Laboratory (INEEL). Funds were provided by the office of Civilian Radioactive Waste Management and Office of Environmental Management. The prototype is based on the Transnuclear design and is being fabricated to quality assurance requirements of the Nuclear Regulatory Commission and Department of Energy Office of Civilian Radioactive Waste Management, i.e., RW-0333P. Procurement has been completed and DTS components should be delivered to INEEL in January 1998. EPRI has contracted with Pacific Northwest National Laboratory for preparation of the demonstration test plan that effort will be coordinated with DTS Utility Advisory Group. Cold tests with a mockup commercial nuclear fuel assembly are scheduled from March 1998 to December 1998. The demonstration will validate system performance, component fabricability, costs, and other parameters. Subsequent hot tests with spent nuclear fuel assemblies, estimated to cost an additional $1 million, are contingent on the availability of funds and the INEEL Test Area North Hot Shop Facility.

The DTS consists of several subsystems, e.g., fuel handling subsystem, and a facility to provide shielding during fuel transfer operations. The equipment was designed by SGN, a subsidiary of Cogema, and incorporates experience gained from French nuclear operations at La Hague. All systems, except the concrete facility, are designed to be portable. The floors and ceilings in the concrete facility, are designed to be portable. The floors and ceilings in the concrete cell that contain the handling equipment lift in-and-out to accommodate use of the same dry transfer system equipment at different locations.

A. Facility

The base dimensions of the facility will be approximately 40 x 60 feet with a height of approximately 45-50 feet. It consists of a preparation area, a lower access area and a transfer confinement area. The preparation area is a sheet metal building where casks are prepared for unloading, loading or shipment. The lower access area and transfer confinement area are the first and second floor, respectively, of the same building. This part of the facility is made of concrete walls approximately 3 feet thick. A cask is moved into the lower access area from the casks preparations area. A large shield door separates the preparation area from the lower access area. The lower access area and the transfer confinement area are separated by a floor containing two portals in which the casks are aligned. The fuel handling machine is located in the transfer confinement area and moves fuel assemblies from one cask to the other. On the roof of the transfer confinement area is a crane dedicated to handling casks shield plugs and lids. The crane can be operated manually for off-normal recovery. Each area of the facility has a separate heating, ventilation and air conditioning (HVAC) system. The HVAC systems are balanced to ensure airflow from the preparation area (uncontaminated) to the lower access area, to the transfer confinement area (potentially contaminated). The control room and HVAC systems are separate from the facility and are envisioned to be portable, i.e., housed in a trailer or van. The transfer operations are performed remotely, however, maintenance on the facility equipment is manual. Robotics are not used because of cost considerations and technical complexity.
B. Fuel Handling Machine

The fuel handling machine is a single failure proof crane. It incorporates a transfer tube that contains the spent nuclear fuel assembly during the transfer operations. At the bottom of the transfer tube is an “anti-falling device” which closes when the spent fuel assembly is in the transfer tube. The anti-falling device prevents crud from falling off the spent fuel during transfer and spreading contamination in the transfer confinement area. When the spent fuel transfer tube is aligned with the receiving cask, the anti-falling device opens and the accumulated crud falls into the receiving cask. There will be a monitoring system in the facility to ensure proper grappling of the fuel. Two systems are planned: (1) a video monitor and (2) a series of switches, to assure that the operator knows the attitude of the fuel at all times. The fuel handling machine can be operated manually from the facility catwalks for off-normal recovery.

The demonstration at INEEL will be conducted in the INEEL Test Area North Warm Shop. A structural steel space frame will support the DTS sub-systems and mockups of the TN source cask and VECTRA MP-187 receiving cask. EPRI and DOE-Idaho will publish final reports.
The “Uranium, ATSDR Public Health Statement” was not available for this draft.
DUAGG/DUCRETE Development Requirements

DUAGG™/DUCRETE™ Experimental Program

July, 1997

PREPARED BY:

LOCKHEED MARTIN IDAHO TECHNOLOGIES COMPANY

and

NUCLEAR METALS, INC.
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Tables

Table 1. Depleted Uranium Inventories in the DOE Complex .............. 1
Uranium

ATSDR Public Health Statement
December 1990

This Statement was prepared to give you information about uranium and to emphasize the human health effects that may result from exposure to it. The Environmental Protection Agency (EPA) has identified 1,177 sites on its National Priorities List (NPL). Uranium has been found above background levels at 26 of these sites. However, we do not know how many of the 1,177 NPL sites have been evaluated for uranium. As EPA evaluates more sites, the number of sites at which uranium is found may change. The information is important for you because uranium may cause harmful health effects and because these sites are potential or actual sources of human exposure to uranium.

When a radioactive chemical is released from a large area such as an industrial plant, or from a container such as a drum or bottle, it enters the environment as a radioactive chemical emission. This emission, which is also called a release, does not always lead to exposure. You are exposed only when you come into contact with the radioactive chemical. You can come into contact with it in the environment through breathing air, eating, drinking, or smoking substances containing the radioactive chemical. Exposure may also result from skin contact with the radioactive chemical alone, or with a substance containing it. Exposure can also occur by being near radioactive chemicals in concentrations that may be found at hazardous waste sites or at industrial accidents.

If you are exposed to a hazardous substance such as uranium, several factors will determine whether harmful health effects will occur and what the type and severity of those health effects will be. These factors include the dose (how much), the duration (how long), the route or pathway by which you are exposed (breathing, eating, drinking, or skin contact), the other chemicals to which you are exposed, and your individual characteristics such as age, sex, nutritional status, family traits, lifestyle, and state of health.

What is uranium?

Natural uranium is a silver-colored metal that is radioactive. Small amounts of uranium are present in rocks, soil, water, plants, and animals and contribute to the weak background radiation from these sources. Soil commonly contains variable amounts, but the average is about 2 parts uranium per million parts of soil (2 ppm). This is equivalent to a tablespoon of uranium in a truckload of dirt. Fertilizers made from phosphate rocks contain higher amounts of uranium than natural soils. Some rocks and minerals in underground and open pit mines also contain uranium in a more concentrated form. After these rocks are mined, uranium is extracted and chemically
converted into uranium dioxide or other usable forms. The remaining rock from which uranium has been extracted is called depleted ore or mill tailings.

Natural uranium is composed of three forms (called isotopes) of uranium: uranium-234, uranium-235, and uranium-238. The amount of uranium-238 in natural uranium is more than 99%. Uranium-235 is present at just 0.72%, in natural uranium, but it is more radioactive (and therefore more hazardous) than uranium-238. Uranium-235 is used in nuclear bombs and nuclear reactors. An industrial process by which the percent of uranium-235 is concentrated is called enrichment, and the uranium obtained this way is called enriched uranium. Uranium-234 is even less abundant than uranium-235, so it can be ignored for most practical purposes.

Uranium-238 is not stable but breaks down into two parts. This process of breaking down is called decay. The decay of uranium-238 produces a small part called “alpha” radiation and a large part called the decay product. The breakdown of uranium-238 to its decay products happens very slowly. In fact, it takes about 4.5 billion years for one-half of the uranium-238 to break down (4.5 billion years is the half-life of uranium-238). Thorium, the decay product of uranium, is also not stable, and it continues to decay until stable lead is formed. During the decay processes, the parent uranium-238, its decay products, and their subsequent decay products release a series of new elements and radiation, including such elements as radium and radon, alpha and beta particles, and gamma radiation. Alpha particles cannot pass through human skin, whereas, gamma radiation passes through more easily.

Because of the slow rate of decay, the total amount of natural uranium in the earth stays almost the same, but it can be moved from place to place through natural processes or by human activities. When rocks are broken up by water or wind, uranium becomes a part of the soil. When it rains, the soil containing uranium can go into rivers and lakes. Mining, milling, manufacturing, and other human activities also move uranium around natural environments.

We use uranium mainly in nuclear power plants and nuclear weapons. Very small amounts are used in making some ceramics, light bulbs, photographic chemicals, and household products.

How might I be exposed to uranium?

Since uranium is found nearly everywhere, you can be exposed to it in the air, water, food and soil. We know, roughly, the average amounts of uranium in food (0.08 to 70 micrograms per kilogram [ug/kg]) and drinking water (0.4 to 1.4 micrograms per liter [ug/L]; 1 microgram = 1/1,000,000th of 1 gram). Most people in the United States take in some uranium with their food every day. Root vegetables, such as beets and potatoes, tend to have a little more uranium than other foods. In a few places, the
concentration of uranium is higher in the water than in the food. People in these areas take in more uranium from their drinking water than from their foods. Your daily intake of uranium may be greater than average if you live near uranium mines or processing plants or an uncontrolled waste site containing uranium, eat food grown in contaminated soil, or drink water that contains unusually high levels of uranium. Normally, very little of the uranium in lakes, rivers, or oceans gets into the fish or seafood we eat. The amount in air is usually so small that it can be safely ignored.

However, people who work at factories that process uranium, work with phosphate fertilizers, or live near uranium mines have a greater chance of being exposed to uranium in the air than most other people. Larger-than-normal amounts of uranium might also enter the environment from accidental discharges from uranium processing plants.

How can uranium enter and leave my body?

If you were to breathe in uranium dust, most of it would leave the lungs when you cough or breathe out. However, you might swallow some of the uranium you breathe in as your body removes the uranium from your lungs. Some of the uranium in your lungs will enter your blood, pass through the kidneys, and be eliminated in the urine within a few days. A small amount may stay in your lungs for years.

Since uranium is present all over the earth, everyone normally eats or drinks a small amount of uranium daily. When it enters your body this way, about 99% of it leaves within a few days in your feces and never enters your blood. A small amount of uranium (about 1%) will enter the blood. Most of this will pass through the kidney and be eliminated in the urine in a few days. A small amount goes to your bones and may stay in your bones for years. A very small amount, about 1/5000th of the weight of an aspirin tablet, is found in most people, mainly in their bones.

Although uranium is radioactive, the type of radiation it gives off cannot go through your skin, so natural uranium that is outside the body is not hazardous. When uranium gets inside your body, after breathing it in, eating or drinking it, or through cuts in your skin, radiation and chemical toxicity are of concern to health.

How can uranium affect my health?

We do not know for certain if natural uranium is dangerous to human health, although evidence of kidney effects were seen in people who work in uranium mines. Animals have developed kidney disease after they have been exposed to large amounts of natural uranium in the food, in the drinking water, in the air, or on the skin.
There is always a concern about getting cancer from any radioactive material. Natural uranium has very low levels of radioactivity and has not definitely been shown to cause cancer in humans or animals. Nevertheless, it is possible that you could develop cancer from swallowing or breathing large amounts of natural uranium because the greater your exposure to a radioactive material, the greater your chance of developing cancer. This is particularly true for enriched uranium that has been made more radioactive. Cancer may develop many years after swallowing or breathing a radioactive material. Just being near natural uranium is of very little danger to your health because most of the radiation given off by uranium cannot go through your skin.

We do not know if natural uranium causes reproductive effects or birth defects in humans, but animal studies suggest that uranium may affect reproduction and the developing fetus.

What levels of exposure have resulted in harmful health effects?

Small amounts of uranium are always in your body and these amounts are not known to affect your health. Some uranium miners have developed lung cancer. This cancer is not from the uranium itself, but from the high levels of radioactive radon gas, which is formed when uranium decays.

Animals that ate food, drank water, or breathed air that had high levels of uranium dust have developed kidney damage. The extent of kidney damage depends on how much uranium gets into their bodies and on the chemical form to which the animals are exposed. Animals can eat or breathe large amounts of some forms of uranium without having any health problems at all.

Tables 1-1, 1-2, 1-3, and 1-4 show the relationship between uranium and known health effects.

Is there a medical test to determine whether I have been exposed to uranium?

There are medical tests that can be performed to determine the amount of uranium in your urine and feces. If you are exposed to a larger-than-normal amount of uranium, some uranium may appear in your urine and feces. Since most uranium leaves the body in the feces within a few days, the amount in the feces only shows whether you have been exposed to a larger amount than normal within the last week or so. Uranium can be found in your urine for up to several months after exposure. The amount of uranium in your urine and feces does not always accurately show how
much uranium you were exposed to.

Since uranium is known to cause kidney damage in humans and animals, urine tests can be used to see if you have kidney damage that may have been caused by exposure to uranium. Some of these tests include measuring the amount of protein, sugar, or enzymes in the urine, or detecting the presence of damaged kidney cells in the urine. These tests, however, are not specific for uranium and are only useful to determine if kidney damage has occurred.

If you breathe large amounts of radioactive uranium, the amount of radioactivity in your body can be measured by a special test. This test is only useful if you have been exposed to certain types of uranium that stay in the lungs for a long time, or to enriched uranium that is more radioactive than normal.

What recommendations has the federal government made to protect human health?

EPA states that long-term exposure to 0.003 milligrams of uranium/kilogram of body weight/day in the food or drinking water is safe for humans. This value is for compounds of uranium that dissolve easily in water. EPA requires industries to report discharges of more than 0.1 curie for most uranium isotopes, including uranium-238, and to report spills of 100 pounds or more of two uranium compounds, uranyl nitrate and uranyl acetate. Uranium levels in the workplace are regulated by the Occupational Safety and Health Administration (OSHA) and recommended by the National Institute for Occupational Safety and Health (NIOSH). Both organizations set the occupational exposure limit for an 8-hour workday, 40-hour workweek at 50 micrograms per cubic meter (ug/m³) for uranium compounds that dissolve easily in water. The limits for compounds that do not dissolve easily in water are 200 ug/m³ (OSHA) and 250 ug/m³ (NIOSH).
TABLE 1-1. Human Health Effects from Breathing Uranium

<table>
<thead>
<tr>
<th>Levels in Air</th>
<th>Length of Exposure</th>
<th>Description of Effects</th>
</tr>
</thead>
<tbody>
<tr>
<td>Short-term Exposure</td>
<td></td>
<td>The health effects resulting from short-term human exposure to air containing specific levels of uranium are not known.</td>
</tr>
<tr>
<td>(Less than or equal to 14 days)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Long-term Exposure</td>
<td></td>
<td>The health effects resulting from long-term human exposure to air containing specific levels of uranium are not known.</td>
</tr>
<tr>
<td>(greater than 14 days)</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
TABLE 1-2. Animal Health Effects from Breathing Uranium

<table>
<thead>
<tr>
<th>Levels in Air (mg/m³)</th>
<th>Length of Exposure</th>
<th>Description of Effects*</th>
</tr>
</thead>
<tbody>
<tr>
<td>630</td>
<td>10 min</td>
<td>Slight kidney damage in rats.</td>
</tr>
<tr>
<td>12,000</td>
<td>10 min</td>
<td>Death in rats.</td>
</tr>
<tr>
<td>18,000</td>
<td>2 min</td>
<td>Slight kidney damage in guinea pigs.</td>
</tr>
<tr>
<td>62,000</td>
<td>2 min</td>
<td>Death in guinea pigs.</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Levels in Air (mg/m³)</th>
<th>Length of Exposure</th>
<th>Description of Effects</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.05</td>
<td>1 year</td>
<td>Slight kidney effects in rats.</td>
</tr>
<tr>
<td>0.20</td>
<td>7.5 months</td>
<td>Kidney damage in guinea pigs.</td>
</tr>
<tr>
<td>0.25</td>
<td>6.5 months</td>
<td>Kidney damage and death in rabbits.</td>
</tr>
<tr>
<td>0.25</td>
<td>1 year</td>
<td>Kidney damage and death in dogs.</td>
</tr>
<tr>
<td>5.0</td>
<td>5 years</td>
<td>Lung damage in monkeys.</td>
</tr>
</tbody>
</table>

* These effects are listed at the lowest level at which they were first observed. They may also be seen at higher levels.
## TABLE 1-3. Human Health Effects from Eating or Drinking Uranium

<table>
<thead>
<tr>
<th></th>
<th>Short-term Exposure (Less than or equal to 14 days)</th>
<th>Long-term Exposure (greater than 14 days)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Levels in Food</strong></td>
<td><strong>Length of Exposure</strong></td>
<td><strong>Description of Effects</strong></td>
</tr>
<tr>
<td></td>
<td></td>
<td>The health effects resulting from short-term human exposure to food containing specific levels of uranium are not known.</td>
</tr>
<tr>
<td><strong>Levels in Water</strong></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

The health effects resulting from short-term human exposure to water containing specific levels of uranium are not known.

The health effects resulting from long-term human exposure to water containing specific levels of uranium are not known.
<table>
<thead>
<tr>
<th>Levels in Food (ppm)</th>
<th>Levels in Water (ppm)</th>
<th>Length of Exposure</th>
<th>Description of Effects</th>
</tr>
</thead>
<tbody>
<tr>
<td>9480</td>
<td>16</td>
<td>1 dose</td>
<td>Rats had fewer pups.</td>
</tr>
<tr>
<td></td>
<td>716</td>
<td>Days 6-15 of pregnancy</td>
<td>Deformities in the pups of mice and weight loss in the mothers.</td>
</tr>
<tr>
<td></td>
<td>820</td>
<td>1 dose</td>
<td>Death in mice</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td>1 dose</td>
<td>Death in rats.</td>
</tr>
<tr>
<td>94</td>
<td>16</td>
<td>30 days</td>
<td>Kidney damage in rabbits.</td>
</tr>
<tr>
<td>469</td>
<td>8-14 weeks (during and after pregnancy)</td>
<td>30 days</td>
<td>Death in rabbits.</td>
</tr>
<tr>
<td>1940</td>
<td>Day 13 of pregnancy to day 21 of nursing</td>
<td>2 years</td>
<td>Death in rats.</td>
</tr>
<tr>
<td>2315</td>
<td>64</td>
<td>4 weeks</td>
<td>Death in mice.</td>
</tr>
<tr>
<td></td>
<td>471</td>
<td>4 months</td>
<td>Decreased weight of pups in mice.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>16</td>
<td>Maternal death in mice.</td>
</tr>
<tr>
<td></td>
<td>21</td>
<td>8-14 weeks (during and after pregnancy)</td>
<td>Kidney and liver damage and blood effects in rats.</td>
</tr>
<tr>
<td></td>
<td>64</td>
<td>4 weeks</td>
<td>Damage to the testes in rats.</td>
</tr>
</tbody>
</table>
A SUMMARY
OF
DOE OFFICE OF ENVIRONMENTAL MANAGEMENT
EFFORTS TO EXPLORE RECYCLE AND REUSE
OF
THE DEPARTMENT’S DEPLETED URANIUM RESOURCE

BY

CARL R. COOLEY, EM-50
Delivered at the DU Workshop
July 15-17, 1997
Las Vegas, Nevada
1. **Background**

**History and Status of DU**

The U.S. Department of Energy (DOE) has been enriching uranium for defense and commercial nuclear fuel purposes since the 1940s. The enrichment process produces large quantities of depleted uranium (DU), i.e., uranium containing less than 0.7% of its $^{235}$U isotope (the assay of most of the DU inventory is 0.2 to 0.5%). Over the years, this DU has been stored at the DOE uranium enrichment sites at K-25 in Oak Ridge, Portsmouth, OH, and Paducah, KY, in 10- and 14-ton steel cylinders, as solid uranium hexafluoride ($^{235}$UF$_6$).

In March 1994, there were 555,000 metric tons of UF$_6$ (equivalent to 375,000 metric tons of uranium metal) stored in 46,000 cylinders throughout the DOE complex. The total amount of the resource is now closer to 600,000 metric tons. Since the United States continues to enrich uranium for commercial nuclear power customers, some 2,500 cylinders are annually added to this inventory. However, since July 1, 1993, uranium enrichment (and, hence, UF$_6$ generation) has been the responsibility of the U.S. Enrichment Corporation (Gary, until the USEC is privatized, it is my understanding that the DOE is still responsible for the UF6. Congress approved such a bill last year.).

The cylinders containing DU are sitting in the open on concrete pads or on gravel beds. Some have been stored for over 40 years. Until recent changes in world affairs, there was no thought of doing anything with the DU other than storing it until it could be used for further enrichment or as fertile blanket material for breeder reactors. Minor quantities of DU are used for tank armor and/or for armor-piercing penetrators on artillery shells. These latter uses continue, but the demand is very small. While the DOE Field Office in Oak Ridge reported that only about a half dozen of the 50,000 UF$_6$-filled containers have leaked, public perception is that these DU containers pose still another risk of a DOE pollution problem. Photographs showing external rusting of the canisters do little to instill public confidence.

During the last several years, environmental activists in and officials from the State of Ohio have posed the question as to whether DU is a waste. DOE's position is that DU is a "source material" under terms of the Atomic Energy Act of 1954, as amended, not a waste, and not subject to the Resource Conservation and Recovery Act (RCRA). Yet, plans for site closure and environmental remediation at Fernald and NTS, respectively, has relatively large quantities of depleted uranium metal, uranium tetrafluoride, and uranium oxide going to disposal.

One exemption from stringent storage and disposal regulations that RCRA requires affords owners of hazardous waste designation of the waste as a "Recyclable Material". However, to meet the definition of "recyclable" there must be a specific beneficial use identified for the material.
EM Work on DU

In the summer of 1992, the DOE Office of Technology Development (now the Office of Science and Technology, OST, EM-50) began to explore whether the large stores of DU could indeed be used for radiation shielding in spent nuclear fuel or high-level waste storage casks, or in other products. In FY 1993, EM-50 initiated a series of studies, using both DOE laboratories and private sector subcontractors, to better understand options available for disposal of the UF₆ and to evaluate possible uses for the depleted uranium. Lastly, since the potential uses of depleted uranium are a strong function of cost (and since the cost of disposing of the material runs into the billions of dollars), other studies were initiated to identify possible technology development opportunities that might reduce these costs. It was intended that these studies provide a basis for establishing future DOE policy and research directions for depleted uranium storage, disposal, or future use. These studies and the respective contractors are listed below:

1) **Depleted Uranium Disposal Options Evaluation [SAIC-Idaho/INEL]**: The study identified and evaluated the viable options for disposal of DU. The conversion of the UF₆ to a stable chemical form, regulatory requirements, and functional/operational parameters were considered in estimating the cost of disposal of the DU as a low-level waste. Additionally, the cost impact on disposal of DU if it were ever declared a RCRA waste was estimated (Reference 2);

2) **HLW Transport and Disposal Evaluation [Sandia National Laboratory]**: The study examined the use of DU metal as shielding material in a storage and transportation cask for DOE's vitrified HLW. The study assessed the cost advantages/disadvantages of several DU HLW cask options throughout the defense high-level waste management program (Reference 3);

3) **Commercial Capabilities Review [Technics Development Corp.]**: A market survey was performed to assess the commercial capabilities and costs for conversion of UF₆ to UF₄ and uranium metal and for fabrication of DU metal components. Additionally, regulatory and strategic issues associated with DU recycling were identified (Reference 4);

4) **Federal Capabilities Review [ORNL]**: A survey of depleted uranium conversion and manufacturing capabilities of the Federal government was performed. The study included identification of areas for manufacturing technology development (Reference 5);

5) **Depleted Uranium Management Alternatives [SAIC-Idaho/INEL]**: The study looked at the costs of two management alternatives for depleted uranium -- continued storage and conversion of UF₆ into metal for use in a spent fuel storage and transportation cask. A preliminary, conceptual special nuclear fuel (SNF) cask design was developed and costs for production estimated (Reference 6);

6) **Development of DUCRETE [INEL]**: Concrete mixtures containing UO₂ fuel pellets were studied (concrete with DU aggregate is called DUCRETE -- see more below). Mixtures of conventional concrete and DUCRETE were fabricated and cured in identical conditions. The
samples were subjected to compression and tensile testing. Results from the testing established the adequacy for spent fuel storage cask applications (Reference 7);

7) Depleted Uranium Concrete Container Feasibility Studies [PACTEC/INEL and SNC/INEL]: Two independent studies were performed to assess the feasibility of using depleted uranium in concrete as a substitute for conventional aggregate. The depleted uranium concrete, DUCRETE, was shown to have tremendous shielding advantages for applications such as spent fuel storage casks. Feasibility was clearly established and major weight reductions and diameter reductions were identified. These studies identified potential significant advantages to the use of a material where the uranium (gamma shielding) and the hydrogen (neutron shielding) can be varied somewhat independently based on the aggregate to cement and water ratio (References 8 and 9);

8) Conceptual Design of A Transportable DUCRETE Storage Cask [SNC/INEL]: This study evaluated the concept of using a DUCRETE dry storage cask inside of a stainless steel overpack for transport to the repository. Although feasible, the design concept is substantially more complex due to heat transfer limitations during transport (Reference 10);

9) Comparative Economics for DUCRETE Spent Fuel Storage Cask Handling, Transportation, and Capital Requirements [SNC/INEL]: This study compared conventional concrete storage casks to DUCRETE casks on a life-cycle cost basis for three time frames and three utility sizes. Significant economies were identified for the DUCRETE storage system when the cask was used for reactor storage and as a disposal overpack in the geologic repository (Reference 11);

10) Depleted Uranium Plasma Reduction System Study [MK/INEL]: This study provided a life-cycle cost estimate for the production of uranium metal with an advanced process that uses a plasma reactor to produce uranium metal and anhydrous hydrogen fluoride gas (Reference 12);

11) Depleted Uranium Market Study (Kaplan of Oak Ridge): This study found that the only large market potential for depleted uranium was shielding applications (Reference 14).

12) Depleted Uranium: A DOE Management Challenge: This report summarized the major results of all the foregoing studies (Reference 1).

2. Options Explored

Alternative Uses and Potential DU Amounts Consumed

Through these investigations, OST explored other possible uses for DU in addition to shielding in canisters or storage casks. After being examined these options, as far as OST is concerned, have been abandoned either due to cost, appropriateness of a Federal environmental R&D
organization to fund, the existence of more appropriate funding organizations, or dubiousness of the option.

They are, without description, as follows together with the anticipated amount of DU resource that would be utilized if such a use were adopted:

1. Shielding for NATO airplane/personnel bunkers -- 0.4 - 1.6 M Metric Tons
2. Penetrators for downhole oil/gas drilling rigs -- 300 -1,000 Metric Tons/yr
3. Collar weights for downhole oil/gas drilling rigs -- 10,000-40,000 MT/yr

3: More Viable Solutions

A. DUCRETE

During investigation of alternative uses of DU, and of different forms of DU for the best performance of spent fuel storage casks made of different forms of DU (usually oxide or metal), OST investigators at the Idaho National Engineering Laboratory (INEL) invented the concept of DUCRETE concrete. This concept entails using aggregate made from DU (DUAGG) for incorporation into a concrete formed into casks/containers. The aggregate is uranium oxide in either U3O8 or UO2 form depending on fabrication parameters and end use. The most attractive aspect of this concept is that, being so dense, the DU permits shielded casks to be made with much thinner walls than traditional concrete casks made for this purpose (with DU’s attendant increased properties for gamma and neutron attenuation), and therefore the weight of the overall cask is lighter. Since the weight of such casks is a concern of utilities regarding their at-reactor handling equipment, the potential posed by this concept warranted further studies.

Indeed, further tests showed that DU aggregate concrete is approximately 68% better at shielding gamma radiation and 12% better at shielding neutron radiation than conventional concrete, enabling wall thickness to be reduced by over 40 inches compared to the standard concrete casks. This also reduces the overall vessel weight by over 35 tons while still having sufficient tensile and compressive strengths to meet standards for a SNF storage cask.

Oxidation tests performed on samples of DUCRETE showed that its performance is excellent at temperatures up to 150° C, and further tests are underway to develop “recipes” for the DUCRETE to increase this temperature. However, requirements for at-reactor dry fuel storage casks do not exceed the 150° C temperature. Further work is also being done to determine the thermal, chemical, and physical stability of the material.

Costs of production of DUCRETE containers are expected to be comparable to those for conventional concrete based casks. Offsetting the higher cost of converting depleted uranium
hexafluoride into depleted uranium aggregate is the cost savings associated with factory fabrication. Based on recent studies by DOE-NE, as part of their Programmatic Environmental Impact Statement Engineering studies, the cost for conversion of UF₆ to U₂O₅ is about $0.92 per kg-U. In recent studies by a commercial firm developing pilot scale aggregate production capability, the estimated production costs of the DU aggregate in large scale production would be between $0.25 to $0.35 cents per pound of aggregate. Thus, the net cost of aggregate (51.6 tons) in a cask sized to hold 21 PWR assemblies is about $65K (the UF₆ to oxide conversion cost is about $34K and the aggregate production is about $31K — as opposed to the comparable figure of $1.5M for a DU cast metal as outlined in Reference 6). Combined with other manufacturing costs, the total fabrication costs including profit is estimated at $214K (not taking advantage of DU disposal credits) or $211/kg U of SNF contained. These costs are clearly competitive with conventional concrete and lower than depleted uranium metal or steel casks. The total life cycle cost of DUCRETE casks is lower than that of conventional concrete storage casks if the DU oxide aggregate is furnished by DOE in lieu of the costs for disposal. NE has recently provided costs estimates for disposal of between $2.14 to $5.56 per kg-U as part of the PEIS studies.

Consequently, the cost of disposal of the uranium contained in such a cask ranges from $79K to $206K. Thus, it appears, on the basis of these cost estimates, to justify the use of DU as DUCRETE casks as an alternative to disposal, since casks are needed for spent nuclear fuel management.

B. DUPOLY

In September, 1996, the Brookhaven Natural Laboratory published a report (Reference 15) summarizing work performed for OST to examine the feasibility of processing and characterization of using depleted uranium in a combined manner with polyethylene (DUPOLY) to form a stable final waste form, which offers another potentially viable recycle option for DU.

DU in oxide powder form was encapsulated in low-density polyethylene using a single-screw extrusion process. DU was oven-dried to remove residual moisture prior to processing. Waste and binder materials were fed by calibrated volumetric feeders to the extruder, where the materials were thoroughly mixed and heated to form a homogeneous molten stream of extrudate. The encapsulated DU, called DUPOLY, was then cooled in cylindrical molds for performance testing and round disks for attenuation studies. DU loadings as high as 90 wt% DU were successfully achieved. A maximum product density of 4.2 g/cm³ was achieved using UO₂, but increased product density using UO₂ is estimated at 6.1 g/cm³.

Additional product density improvements up to about 7.2 g/cm³ are estimated using a hybrid technique known as micro/macroencapsulation. Waste form performance testing included compressive strength, water immersion and leach testing. Compression test results are in keeping with measurements made with other waste materials encapsulated in polyethylene. Leach rates were relatively low (0.08 - 1.1%) and increased as a function of DU loading. However, considering the insolubility of uranium trioxide, these leach data indicate the probably
presence of other, more soluble uranium compounds. Ninety day water immersion tests showed sensitivity to one type of UO₃ ("batch" processed) for samples containing > 85 wt% of the oxide. Samples containing UO₃ produced by a newer "continuous" process showed no deterioration at up to 90 wt% DU loadings.

At these densities, the material appears to be very suitable for LLW shielding applications where the source term is greater than the contact handling limits. Polyethylene has been shown to be resistant to radiation damage for exposures up to 10⁶ Rad. However, for SNF and HLW where the surface dose is about 10⁴ R/h, the exposure limit would be reached in a very short time (~1 year) and radiation damage to the polyethylene would preclude its use.

4. Conclusion

Based on existing technology and its associated costs, the minimum cost to DOE of dealing with its current DU inventory is in the neighborhood of $3.9 billion based on earlier EM studies (for conversion to an oxide and subsequent disposal as low-level waste). More recent NE studies have lowered this cost to between $0.8 to $2.1 billion by working with industry to lower the cost of UF₆ conversion compared to the historical values used in the earlier EM studies. This assumes that UF₆ is not declared a RCRA waste and that no special permits are required for continued interim storage in existing cylinders. If the UF₆ were ever declared a RCRA waste, costs for future processing and disposal could more than triple.

The lowest cost option appears to be recycling DU to useful product such as DUCRETE spent fuel and HLW casks. Both the lower cost of converting UF₆ to an oxide and the low cost of fabricating concrete components make these estimated costs lower than those of the metallic uranium casks considered. Although the cost for fabrication of the depleted uranium aggregate is uncertain, there are precedents for large scale, low-cost manufacturing of non-nuclear ceramic materials. DUCRETE concrete casks may not be cost competitive with conventional concrete. However, if DOE provides the converted oxide and DUAGG in lieu of having to pay the cost of disposal, DUCRETE casks then appear very competitive to conventional concrete casks.

Metal uranium storage casks, which have some performance advantages over DUCRETE casks, appear to be much more costly than DUCRETE casks. However, there is a large potential savings in conversion of the UF₆ to uranium metal through application of the direct plasma reduction processes being explored by both INEL and LANL. The technology would directly reduce UF₆ to metal without requiring a reductant such as Mg and, consequently, would eliminate the MgF₂ waste stream and associated costs of MgF₂ disposal. However, this concept has not yet been proved beyond lab scale. To reduce the costs of DU metal casks to levels competitive with current concrete casks or to the projected cost of DUCRETE casks, considerable reduction in the fabrication cost of DU metal is also required. In summary, for DU metal to be competitive, its costs will have to be reduced well below that for present metallic steel and lead casks. This cost objective appears ambitious.
5. References


This paper outlines the sequence of events and associated work required to advance the DUAGG™/DUCRETE™ program into the production of licensed, spent nuclear fuel casks. The principle of forming an artificial aggregate using various compounds of uranium oxide (UO$_2$, UO$_3$, and U$_3$O$_8$) undergoing a liquid phase sintering process have been successfully demonstrated at the Idaho National Engineering and Environmental Laboratory (INEEL). Future work needs to include: (1) additional material properties testing in support of cask design data requirements, (2) depleted uranium concrete (DUCRETE) manufacturing studies, (3) cask design and review, (4) cask construction and testing, and (5) Safety Analysis Report (SAR) in support of storage license and Certificate of Compliance.

The most important work needing to be performed, in the near term, is DUAGG and DUCRETE materials property testing and manufacturing studies. These two activities can be performed concurrently and together provide the most critical need for cask design, construction, and testing.
1. Introduction

Depleted uranium inventories are found throughout the DOE Complex as well as some DOD facilities. Some private supplies of DU also exist but to a much smaller extent. Most (95%) of the DOE inventory exists in the form of uranium hexafluoride - UF₆. Other smaller but still significant quantities (>50 million pounds) of depleted uranium exists at DOE sites in other chemical forms. Table 1 below identifies the major known quantities and locations.

Very little of this depleted uranium material has a known future use.

This overall DU inventory situation requires integrated management or large and valuable material inventories may be wasted. If declared a waste, large liabilities may be incurred.

The Idaho National Engineering and Environmental Laboratory (INEEL) in partnership with private industry (Nuclear Metals, Inc.) is developing a method to produce high-density aggregate (artificial rock) primarily consisting of depleted uranium oxide suitable for use as a component of high-density concrete used as radioactive shielding.

<table>
<thead>
<tr>
<th>Location</th>
<th>Chemical Form</th>
<th>Quantity (MT)</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>Paducah</td>
<td>UF₆</td>
<td>*</td>
<td>28,351 full cylinders (10 and 14 tons each)</td>
</tr>
<tr>
<td>Portsmouth</td>
<td>UF₆</td>
<td>*</td>
<td>13,388 full cylinders (10 and 14 tons each)</td>
</tr>
<tr>
<td>Oak Ridge</td>
<td>UF₆</td>
<td>*</td>
<td>4,682 full cylinders (10 and 14 tons each)</td>
</tr>
<tr>
<td>Fernald</td>
<td>Umetal</td>
<td>2000</td>
<td>Planned for disposal in FY1998</td>
</tr>
<tr>
<td>Nevada Test Site</td>
<td>Umetal</td>
<td>70</td>
<td>Sheet metal scrap, Planned for disposal by September 1997</td>
</tr>
<tr>
<td>Savannah River</td>
<td>Umetal</td>
<td>950 1600</td>
<td>Bare metal/Nickel plated-aluminum clad</td>
</tr>
<tr>
<td>Fernald</td>
<td>UF₄</td>
<td>2000</td>
<td>Planned for disposal in FY1998</td>
</tr>
<tr>
<td>Savannah River</td>
<td>UO₂</td>
<td>19420</td>
<td>Indefinite Storage</td>
</tr>
<tr>
<td>Fernald</td>
<td>UO₂</td>
<td>40</td>
<td>Planned for disposal in FY1998</td>
</tr>
</tbody>
</table>

* Total inventory at these sites is generally estimated to be 550,000 MT as of the transfer of the facilities to the US Enrichment Corporation in July 1993. Upon privatization, material generated since transfer (~2000 cylinders per year will also revert to DOE).
The oxide materials identified in Table 1 are sufficient to fabricate large quantities of depleted uranium concrete (DUCRETE®) containers for LLW, HLW or spent fuel storage over packs. The Savannah River UO₂ inventory is sufficient to fabricate over 500 spent fuel storage over packs.

Work to date has focused on developing and demonstrating the viability of producing depleted uranium aggregate (DUAGG®). Developmental testing has shown that DUAGG can be produced with suitable chemical and mechanical properties.

The shielded container application will require additional testing, beyond developmental testing, for validation of mechanical, chemical, and nuclear properties of both the DUAGG and depleted uranium concrete (DUCRETE®) materials. Additionally, manufacturing processes need to be optimized to reduce depleted uranium aggregate production costs.

2.0 Technical Needs

This document describes activities related to the maturation of DUCRETE technology which need be performed. Also included is the estimated time needed to resolve the issues under investigation.

2.1 Aggregate Production

For depleted uranium recycle to be cost effective, the process cost for converting UO₂ into aggregate (DUAGG) needs to be inexpensive and environmentally acceptable.

The conversion will be performed in two steps:

1.) Chemical conversion of the UF₆ to Oxide, and

2.) Production of aggregate from the oxide.

Processes have been identified for these conversion steps, however, process optimization is needed to verify that performance is maximized and costs are minimized.

2.1.1 Aggregate Production Optimization Studies

The INEEL has developed the bench scale process techniques for making DUAGG depleted uranium aggregate. Production of aggregate in any scale beyond a few kilograms has not yet occurred.

Nuclear Metals, Inc. (NMI) has obtained a license for production of DUAGG from Lockheed Martin Idaho Technologies Company (LMITCO). NMI is installing pilot scale equipment for DUAGG production in its Carolina Metals facility located near Barnwell, SC. NMI has contracted with LMITCO for production of 6 tons of DUAGG for use in subsequent test programs. Initial production will follow the process steps developed at INEEL.

Future work should involve experimenting with alternative processing steps to simplify processes and lower total production costs.

The current process has potential for cost reduction through simplification.

For example dry versus wet grinding will save considerable energy and reduce the number of processing steps. Wet grinding was chosen to simplify contamination control. If dust control can be accomplished in the grinding operation, reduced operating costs can be assured. Alternative processes for reduction of UO₂ to UO₂ should also be investigated to reduce the sintering cycle time.

2.1.2 Economic Analysis

Most of the current cost estimates are based on simple conceptual designs of a
processing facility or upon processing concepts which now are nearly two years old. For establishing the incentive for process optimization and for developing the absolute cost for DUAGG production, a detailed facility design should be developed and life cycle costs estimated. Such a facility should be designed at a couple of capacity throughputs to meet the potential demand of different recycle scenarios.

2.2 DUCRETE Product Development

DUCRETE Concrete is proposed as a solution to an excess material problem. DUCRETE can be used as nuclear shielding but classic demand pull market forces will not cause customers to demand DUCRETE Shielding for their shielding needs. Most of the time, other traditional materials will work just fine and both costs and engineering parameters are better understood. Thus, nuclear engineers, being risk adverse, will stay with known materials. Consequently, to help the user community view DUCRETE as a viable solution to their shielding needs, DOE must create products and demonstrate their effectiveness. This section discussed data manufacturing needs and product demonstrations which will provide the technical basis for wide scale use of DUCRETE Concrete.

2.2.1 DUCRETE Manufacturing Technologies

No DUCRETE Concrete has been fabricated other than the initial work done as a proof of concept in 1994. This work used surplus depleted uranium oxide fabricated as reactor fuel pellets. These fuel pellets were far from an ideal aggregate due to both their size, shape and surface finish. Nevertheless, based on the results of compression tests, the concrete samples made from them clearly showed that DUCRETE concrete was feasible. However, since those initial tests, no additional DUCRETE concrete has been made.

The high density DUAGG aggregate is the basis behind the enhanced performance attributes of DUCRETE shielding. Most traditional concrete weighs about 140 lb/ft³. Theoretical estimates of DUCRETE concrete are near 450 lb/ft³.

At this density, several new issues need to be addressed:

1) Mixing methods,
2) Fabrication techniques,
3) Mixture stiffness in the wet state,
4) Optimization of super-plasticizer usage,
5) Control over aggregate settling, and
6) Aggregate size range in the mixture.

Methods to pour large concrete structures need to be tested and process steps developed. Mixing processes versus using pre-placed concrete methods need to be evaluated. Aggregate size fractions must be established for prevention of separation. Aggregate mixture densities need to be analyzed to assure homogenous distribution of aggregate. Samples need to be made, tested, and process modification identified as required.

2.2.2 DUCRETE Concrete Material Properties for Structural Applications

Once concrete mixtures have been reasonably established, additional mechanical, thermal, and nuclear property testing, supplementing development testing already performed, must be conducted. The experimental data needs to be acquired
under an approved Quality Assurance Program to establish the needed pedigree for subsequent use in licensing calculations and safety analysis reports. Test data should be acquired for two aggregate loading ratios (50% and 58% volume loading ratio). These additional or supplemental DUCRETE tests include:

**Mechanical Properties**

1) Unconfined compressive strength and tensile strength,
2) Ultimate tensile strength,
3) Density,
4) Bulk modulus (as a function of confining pressure),
5) Stress-strain constitutive relations,
6) Young's modulus, and
7) Poisson's ratio.

**Thermal Properties**

1) Thermal emissivity,
2) Thermal conductivity (0°C to 200°C),
3) Heat capacity, and
4) Coefficient of thermal expansion.

Additional nuclear properties measurements are necessary to validate development testing data and calculated gamma and neutron attenuation characteristics.

Conservative values will be needed for the design of any licensed spent fuel containers. These properties must be obtained in a statistically defensible quantity so that subsequent product design can use the information and support license applications.

### 2.2.3 DUCRETE High Activity LLW Containers

One application for DUCRETE shielding is expected to be for high activity low level waste containers needed for the storage of nuclear materials at DOE sites. To stimulate the use of DUCRETE containers, prototypes must be fabricated and the advantages demonstrated. After successful deployment, potential customers will be more convinced of the material effectiveness. Two potential applications for such containers have been identified at Fernald (~2000 rectangular Type A boxes) and at INEEL (~675 cylindrical drum over packs). Together, these applications will consume 18 million pounds of uranium oxide. This task will develop detailed designs, fabricate prototype containers, test the containers, and develop detailed cost estimates for production quantities.

### 2.2.4 DUCRETE Spent Fuel and HLW Storage Casks

The largest potential application for DUCRETE shielding involves the storage of spent nuclear fuel for DOE Test Reactor Fuel, US Navy Submarine Reactor Fuel, Commercial Power Reactor Fuel, and High Level Waste from the DOE Vitrification facilities. Efforts need to be undertaken to develop (design and manufacture) and test a full scale reactor HLW or spent fuel storage cask.

### 2.3 UF₆ Conversion to Stable Materials

The UF₆ stored at the gaseous diffusion plants was an excellent material for the cascade operations but represents a very poor compound from a long term management perspective. UF₆ is a solid at storage
temperatures but sublimes at slightly elevated temperatures. Upon contact with water, it react to form HF gas and UO$_2$F$_2$, a water soluble solid. The carbon steel cylinders that hold the UF$_6$ have heavily oxidized due to outdoor storage. Consequently, some conversion process will ultimately be needed to convert UF$_6$ to a more stable final form such as uranium oxide (UO$_3$, U$_2$O$_8$ or UO$_2$). Conversion technologies haven't changed much over the last 40 years. Virtually all proven conversion processes involve some elevated temperature reaction of UF$_6$ with steam. Most reactions produce uranium oxide and aqueous hydrogen fluoride. Ideally, a the attributes of the conversion process will minimize the cost, minimize the waste generation, and maximize the value of any product residuals. Even though research needs to be conducted in this area this white paper will not address any conversion processes currently in use or proposed.

3.0 Summary

It is estimated that completion of the additional material testing can be completed in two years with cask design and cask testing completed in four years. This effort will develop the technical basis for the economical production of DUAGG, establish its chemical stability, develop DUCRETE technology, and demonstrate the technical performance of products fabricated from such materials. This technical basis will provide the necessary knowledge and the experience using this new material for deployment and licensing of nuclear shielding systems.
Status Report

DUAGG™/DUCRETE™ Experimental Program

July 1997

PREPARED BY:

LOCKHEED MARTIN IDAHO TECHNOLOGIES COMPANY
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Executive Summary

This report summarizes the development and testing performed, to date, in support of the experimental program to develop a low-cost, artificial mineral containing an extremely high fraction of depleted uranium oxide. Efforts to demonstrate the viability of forming an artificial mineral using various phases of uranium oxide (UO₂, UO₃, and U₃O₈) have shown that a stable aggregate can be formed using a liquid phase sintering process. Testing to evaluate the characteristics of depleted uranium aggregate (DUAGG™) and depleted uranium concrete (DUCRETE™) has included; (1) strength testing, (2) leachability, and (3) oxidation testing. Additional data has been collected in order to establish the parameters for the manufacture of DUAGG, these data include; (1) differential thermal analysis, (2) thermal gravimetric analysis, (3) aggregate density analysis, and (4) metallography testing.
1. Introduction

The Idaho National Engineering and Environmental Laboratory (INEEL) are developing a method to produce high-density aggregate (artificial rock) primarily consisting of depleted uranium oxide. The objective is to develop a low-cost method whereby uranium oxide power (UO₂, UO₃, or U₃O₈) can be processed to produce high-density aggregate pieces having physical properties suitable for use as a component of high-density concrete used as radioactive shielding.

The method being developed utilizes finely divided power consisting of one phase of uranium oxide and increase its packed density via pressing followed by heating (liquid phase sintering technique).

During FY-95 initial demonstrations showed that dense depleted-uranium pellets (produced by traditional methods for use in nuclear reactors - UO₂ pellets) could be used as large aggregate to replace conventional gravel in concrete. The resulting concrete was named DUCRETE™. This DUCRETE was shown to have compression strength levels equivalent to typical construction-grade concrete at room temperature (3,000 to 5,000 psi). Visual examinations found no deleterious interactions between the UO₂ pellet aggregate and the cement/sand/water matrix when cured for seven days at room temperature. Analysis of limited strength data, using only two samples per data point, showed the possibility of a slight degradation in compressive strengths for temperatures between 90°C and 150°C for cure times up to 28 days. Additional testing, of both reference concrete samples (fabricated with gravel aggregate) and UO₂ pellet concrete samples, at temperatures of 250°C, showed decreases in the mean compressive strength of 1,300 psi for the reference sample and either cracking or complete disintegration for the UO₂ pellet concrete samples. It should be noted that this temperature (250°C) is at least 100°C higher than the maximum temperature expected for anticipated applications for actual DUCRETE as spent nuclear fuel storage containers.

During FY-96, the composition optimization for depleted uranium aggregate DUAGG was conducted. The objective of this work was to optimize the DUAGG chemical composition with respect to several parameters: (1) Density, (2) Microstructure (fine grained with minimum of large porosity), (3) Leach resistance (primarily for uranium but also any other elements that might cause degradation of mechanical properties of could be coupled to uranium leaching), and (4) Neutron and gamma ray attenuation when the DUAGG is incorporated into portland cement to form DUCRETE.

The optimized composition was adapted around a compromise on the series of parameters listed above. Specifically, chemical additives that stimulate a high density or neutron attenuation (e.g., boron) were chosen over those additives that optimize leach resistance. It was assumed the primary use of the DUAGG would be for use in concrete (DUCRETE) to ultimately be used as a shielding container for spent nuclear fuel or radioactive wastes. In this application, the DUAGG will be encapsulated in cement to form concrete and in turn the concrete will be contained in a metal (e.g., stainless steel) outer and inter structure such that the DUCRETE would not be directly exposed to rain or ground water.
Therefore, it was assumed that properties like density (good gamma ray attenuation) and the use of a boron additive (good neutron attenuation) has a higher desirability than properties that lead to optimized leach resistance.

The objective of work in 1997 is to develop automated processing techniques that can be scaled-up for mass production using depleted UO$_3$ powder available from the large stock of material stored at Savannah River.

2 Test Results

Test results reported include those tests performed on either UO$_2$ pellet aggregate concrete or optimized DUAGG. The most current data/test results will be presented along with an identifier noting the material used as the aggregate (DUAGG or UO$_2$ pellet aggregate). Also any preliminary data/results from ongoing testing will be presented and flagged. Complete test reports$^3$ can be obtained from the INEEL depleted uranium project office.

2.1 FY-95 Test Results

Samples of DUAGG were fabricated "by hand" using liquid phase sintering of depleted UO$_3$. For samples labeled "SMC" the UO$_2$ starting powder was fabricated at INEEL via oxidation of depleted uranium metal. This powder was milled by hand, using a mortar and pestle for approximately 1/2 hour in order to break up large agglomerates. Various basalt compositions were used to create a liquid phase during high temperature densification (sintering) of the UO$_2$ powder into DUAGG. The samples were fired in a reducing (hydrogen containing) atmosphere using a small "batch" furnace in order to maintain UO$_2$ as the primary phase.

Eighty volume percent UO$_2$ was used in conjunction with twenty volume percent basalt. Samples with reduced iron oxide content were fabricated because leach tests on previous samples indicated that even though uranium leachant remained below the detection limit, iron leachant levels began below the detection limit after two hours of leaching but steadily increased to 1.661 ppm after seven hours to 12.62 ppm after 120 hours (Sample SR2B). The concern was that the integrity of the grain boundary phase could be adversely affected if enough iron leached out when exposed to water over a long time.

In some samples, nine weight percent B$_2$O$_3$ was added to the basalt compositions for the purpose of aiding neutron attenuation. Very efficient in attenuating neutrons (high thermal neutron absorption cross section of 750 barns).

The compositional variables and sintered densities for the UO$_2$ samples are given in Table I. Previous X-ray diffraction analysis has shown that more than 90% of the starting UO$_3$ powder was converted to UO$_2$ after sintering at 1250°C in a reducing atmosphere.
Table I. Sintering of Fine UO₂ powder + Various Compositions of Basalt

<table>
<thead>
<tr>
<th>Sample #</th>
<th>U₀ₓ content (vol. %)</th>
<th>Type of Clay Additive</th>
<th>Description of Mill Additions</th>
<th>Bulk Density (g/cc)</th>
<th>Apparent Density (g/cc)</th>
<th>Open Porosity (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>CPP10</td>
<td>80</td>
<td>J89</td>
<td>Full iron (IEB)</td>
<td>8.59</td>
<td>8.82</td>
<td>2.78</td>
</tr>
<tr>
<td>SRP1A</td>
<td>80</td>
<td>Allen</td>
<td>Full iron (IEB)</td>
<td>6.04</td>
<td>8.02</td>
<td>0.28</td>
</tr>
<tr>
<td>SMC1A</td>
<td>80</td>
<td>Allen</td>
<td>No iron (IEB)</td>
<td>7.59</td>
<td>8.53</td>
<td>0.11</td>
</tr>
<tr>
<td>SMC2A</td>
<td>80</td>
<td>J89</td>
<td>No iron (IEB)</td>
<td>8.18</td>
<td>8.43</td>
<td>0.03</td>
</tr>
<tr>
<td>SMC3A</td>
<td>80</td>
<td>Allen</td>
<td>1/2 iron (IEB)</td>
<td>7.99</td>
<td>8.40</td>
<td>0.49</td>
</tr>
<tr>
<td>SMC4A</td>
<td>80</td>
<td>J89</td>
<td>1/2 iron (IEB)</td>
<td>7.81</td>
<td>7.89</td>
<td>0.01</td>
</tr>
<tr>
<td>SMC5A</td>
<td>80</td>
<td>Allen</td>
<td>Boron-no iron</td>
<td>8.07</td>
<td>8.46</td>
<td>0.05</td>
</tr>
<tr>
<td>SMC5B</td>
<td>80</td>
<td>Allen</td>
<td>Boron-no iron</td>
<td>8.05</td>
<td>8.47</td>
<td>0.05</td>
</tr>
<tr>
<td>SMC6A</td>
<td>80</td>
<td>Allen</td>
<td>Boron-1/2 iron</td>
<td>8.21</td>
<td>8.47</td>
<td>0.03</td>
</tr>
<tr>
<td>SMC6B</td>
<td>80</td>
<td>Allen</td>
<td>Boron-1/2 iron</td>
<td>8.18</td>
<td>8.45</td>
<td>0.03</td>
</tr>
</tbody>
</table>

Note: CPP= U₀ₓ powder obtained from INEL's Chemical Processing Plant
SRP= U₀ₓ powder (-325 mesh) from Savannah River
SMC= U₀₂ powder (primarily) from INEL's Special Manufacturing Capability at (TAN site)
2.1.1 Leachability

Leach testing was conducted in strict adherence to ANSI/ANS-16.1-1986, "Measurement of the Leachability of Solidified Low-Level Radioactive Wastes by a Short-Term Test Procedure."

An Inductively Coupled Plasma (ICP) instrument was used to analyze the leachate solutions. This instrument has a lower detection limit of 0.5 ppm (including uranium). Leach analysis data showed that extremely low levels of uranium are being leached out of all the DUAGG samples. All the DUAGG samples seemed to be remarkably leach resistant with respect to all ions. Sample SR1A (Savannah River starting powders with full iron loading) did show a low level of uranium ions in the leach (1.350 ppm at 120 hr and 0.5363 ppm at 43 days). However, all the remaining DUAGG samples showed uranium ion levels either at 0.0000 or below the detection limit except in a few rare analyses (like SMC4A-24 hr; U= 9.2320 ppm) that probably are due to measurement errors. This result demonstrates DUAGG would be a very leach resistant waste form for uranium oxides.

2.1.2 Microstructure

After sintering, samples were encased in cast plastic holders and subsequently ground and polished down to a final finish. The smallest grit used was a 1 micron diamond paste. After polishing the samples were chemically etched to decorate grain boundaries and an optical microscope used to take photographs at magnifications up to 2,000 X (using an oil immersion lens).

2.1.3 Density

Table I shows that the best densities achieved with UO$_3$ powder from the Idaho Chemical Processing Plant (Sample CPP 10) had a bulk density of 8.59 g/cc and apparent density of 8.82 g/cc. However, samples fabricated using SMC powder (nominally UO$_3$) all gave good bulk densities (7.59 g/cc to 8.32 g/cc) and apparent densities (7.89 to 8.53 g/cc). (SMC3A).

2.1.4 Strength Tests

Samples for strength testing were formed by placing the wet concrete into thick-walled plastic molds. Specimens of the following types were fabricated:

- 2 in. Diameter x 4 in. High for compression testing.
- 2 in. Diameter x 2 in. High for “Brazilian” diametral tensile testing.

Due to the limited amount of sintered UO2 pellets available, only two samples of each type were cast for a given concrete composition.
The samples were cured according to ASTM Standard C 192, *Method of Making and Curing Concrete Test Specimens in the Laboratory*, i.e., held at room temperature in water for the desired time (7, 28, or 90 days).

Sulfur end caps were cast onto the 2 in. diameter x 4 in. high compression specimens before testing to provide flat, parallel ends for testing. This was done according to ASTM Standard C 192, *Method of Making and Curing Concrete Test Specimens in the Laboratory*, and C 617, *Practice for Capping Cylindrical Concrete Specimens*.

The concrete samples were broken using an Instron test machine. The 2 in. diameter x 4 in. high specimens were tested in compression according to ASTM C 39-72, *Compressive Strength of Cylindrical Concrete Specimens*.

The 2 in. diameter x 2 in. high specimens were tested using the Brazilian test. For this test, the disks are placed on an edge and loaded in compression and the failure load, $P$, noted. The splitting tensile strength $S_T$ is given by:

$$S_T = \frac{2P}{\pi td}$$

where

- $t = $ thickness of the disk
- $d = $ disk diameter

The strength values were averaged over only two specimens (due to shortage of UO2 aggregate) and thus should only be taken as a rough indicator of strength trends; a study with more data points would be preferred. All of the 7-day cured mixtures, including the DUCRETE, showed compression strengths that were about the same as those of ordinary construction grade concretes (3,000 to 6,000 psi). The strengths consistently decreased with increasing aggregate content. The UO$_2$-containing mixtures consistently showed slightly lower strengths than the equivalent gravel samples. The difference was smaller at high aggregate loadings. This is probably related to the smooth, regular shape of the UO2 aggregate.

Close examination of the tensile fracture surfaces showed less interlocking and bonding of the sand/cement matrix to the UO$_2$ aggregate than to the gravel. It was shown that when the long axis of the UO$_2$ pellet was perpendicular to the advancing crack, the UO$_2$ pellet would fracture; when the pellet was oblique or parallel to the crack, the fracture was along the interface. There was less of a tendency toward this type of cracking morphology with the gravel aggregate, where the fracture was smoother and a higher fraction of the gravel was fractured.

The tensile test values were approximately 10% of the compression values. The tensile strength of normal concrete is about 8% to 12% of the compressive strength.
2.1.5 High-Temperature Exposure Tests

The high-temperature exposure study was considered to be an "oxidation" study because of the easy access of air and water vapor to the aggregate through the open pores of the cement. There was water vapor available from water not reacted to form calcium silicate hydrate compounds and from water vapor in the atmosphere. The tests were only expected to indicate general trends, since usually only two samples were broken per data point because there was not sufficient depleted uranium oxide aggregate to fabricate more samples. In addition to the small number of duplicate samples, ceramics are brittle materials and typically have a broad strength distribution curve due to a large variation in flaw sizes. As a result, the standard deviation values for the tests are quite variable and sometimes very high.

For the reference (gravel aggregate) concrete, the data indicates a general increase in the compression strength for temperatures up to 150°C and then a decrease in strength at 250°C. The 95% confidence envelope (curved lines) indicates that it is possible that the dependence could actually be flat (no change in mean strength up to 150°C) or very negative (if we had included the 250°C data point in the regression analysis).

Test data for compressive strength of concrete fabricated with depleted UO₂ pellets used as aggregate indicates a slight decrease in strength from room temperature to 150°C. However, the 95% confidence level envelope shows the data could also support a flat dependence (no degradation of strength). The 250°C data were not used because the concrete crumbled. It was clear that a 250°C exposure for 14 days greatly banned both the UO₂ and U₃O₈ aggregate concrete. The effect was greater for the UO₂ than for the U₃O₈. (Maximum long-term exposure temperatures for the concrete in spent nuclear fuel storage containers are expected to be 50 to 100°C.)

It is possible that the tensile data is primarily a measure of the strength of the cement phase (cement plus sand), while the compression data are more of a reflection of the bond between the cement phase and the large aggregate.

Composition SMC5A/5B (No iron, Boron added - of Table I) was selected for use in further experiments. The feed powder used was taken from UO$_2$ (hydrated) powder stored in 55 gallon barrels at Savannah River Site (SRS). This powder was significantly different from the previously used SRS powder. The new powder was highly hydrated and had not been ground and sieved (as with previously supplied “SRS” powder). The experiments conducted in FY 1997 have been to develop process technologies suitable for large scale production of DUAGG.

3.1 Calcination and Milling

The UO$_2$ powder (SRPH) used in FY-97 was first calcined to 900 °C to removed hydrated water. The powder then was milled. Attrition milling was chosen as a very efficient method of grinding the UO$_2$ powder and mixing in the added powders (i.e., calcined and ground soil, clay, calcined mill additions). When compared to vibratory and conventional ball milling, attrition milling is capable of producing the finest product for a fixed specific energy input $^1$.

3.2 Drying and Low Temperature Calcination

The ground and mixed slurry is pumped from the mill into shallow stainless steel containers and dried in a forced convection oven. After drying, the stainless steel pans are transferred into the calcining oven (air atmosphere). This was done to remove about 80% of the chemically absorbed water on the UO$_2$. Care was taken to not remove the hydrated waters from the kaolin clay.

3.3 Binder Addition/Agglomeration

The binder selected for initial testing was “Flamsperse”$^2$, primarily on the basis of low cost and its known use in briquetting. “Flamsperse” is calcium lignosulfonate (a byproduct of the wood pulp industry). It has proven to be an effective processing aid in a variety of ceramic applications.

During laboratory development it has been noted that there seems to been a tendency for the thick hand pressed disks to crack during firing. Several factors are suspected:

1. The thick disks have density gradients due to non-homogeneous stresses during uniaxial pressing (die body was not “floated”).

2. The powders were ground extremely fine inhibiting gases from exiting the body if the binder burns out quickly over a narrow temperature range and causes internal pressure.

3. The binder was added very inhomogeneously using a hand dropper followed by tumbling in the “Turbula”.

The cracking appears to happen early in the firing process (e.g., during binder burnout), thus the cracks are quite wide after firing. This cracking may actually be an advantage, if the DUAGG is to be crushed to provide for a wider distribution of large agglomerate diameters (for a higher strength DUCRETE). However, if cracking proves to be a problem, evaluation of other binders could be pursued.
Limited investigations at the INEEL reveal that three methods appear to be acceptable for adding the binder (in a water solution) that concurrently provides for a good dispersion:

1. Disk Agglomerator
2. Double-cone blender with integral dispersion bar and
3. Pin mixer (e.g., Turbulator or Turbulizer).

### 3.4 Pressing

For developmental work, samples were dry pressed in a metal die (by hand using a uniaxial hydraulic press) to make disks (0.750" dia. x = 0.375" high). Pressures of about 10,000 psi were used.

For production-scale pressing, INEEL has selected briquetting as the lowest cost method for pressing.

### 3.5 Drying

Drying of the briquettes is necessary after briquetting in order to increase the strength and prevent steam from cracking the briquettes during sintering.

### 3.6 Sintering

For developmental tests, the disks were fired (sintered) in a reducing (4% hydrogen in argon) atmosphere in order to maintain UO₂ as the primary phase. Previous X-ray diffraction analysis has shown that more than 90% of the starting UO₂ powder will be converted to UO₂ after sintering at 1250°C in a reducing atmosphere.

About 10 kg of DUAGG has been produced by hand pressing, to date. It takes about 906 g of DUAGG to produce a 2" diameter x 4" long DUCRETE test sample.

### 4.0 DUAGG Product Characterization

#### 4.1 Density

The sintered density characterization of the DUAGG (boron addition - no iron) ground and mixed using the attrition mill is shown in Table II. The binder was added by hand and mixed in with the "Turbula". These bodies showed some cracking (attributed to binder burnout problems), but the sintered densities were quite good.

Samples fabricated without binder addition were difficult to remove from the pressing die and consequently showed some crumbling at the edges. However, their bulk densities were very good and their apparent density values increased significantly over the samples with binder added by hand, while the open porosity decreased. This result indicated that the binder was distributed poorly and leaving residual "pits" or pores within the ceramic microstructure. Results are shown in Table III.

### 4.2 Microstructure

The microstructure of DUAGG has dramatically improved through the use of the attrition mill to better grind and mix the reactant materials.
This improvement is documented in Figure 1 through Figure 3. Figure 1 shows the microstructure fabricated using hand mixing (mortar and pestle) and hand addition of the binder. Figure 2 shows the microstructure fabricated using attrition milling and hand addition of the binder. Figure 3 shows the microstructure fabricated using attrition milling with no binder addition.

All of the disks were uniaxially pressed in a metal die (as previously described, above) then sintered in the batch furnace. The use of an automated method (one of the three described, above) to add the binder should enable the ceramic to be pressed to the densities typical of Table III with microstructures similar to those shown in Figure 3.

4.3 Attenuation

Attenuation testing is presently being performed. Test data not available at this time.
Table II. Sintering of Attrition Milled UO₂ Powder
(Binder Added by Hand)

<table>
<thead>
<tr>
<th>Sample #</th>
<th>UO₂ content (vol. %)</th>
<th>Type of Clay Additive</th>
<th>Description of Mill Additions</th>
<th>Bulk Density (g/cc)</th>
<th>Apparent Density (g/cc)</th>
<th>Open Porosity %</th>
</tr>
</thead>
<tbody>
<tr>
<td>SRPH-1</td>
<td>80</td>
<td>Allen</td>
<td>Boron- No iron</td>
<td>8.01</td>
<td>8.13</td>
<td>1.46</td>
</tr>
<tr>
<td>SRPH-2</td>
<td>80</td>
<td>Allen</td>
<td>Boron- No iron</td>
<td>8.15</td>
<td>8.16</td>
<td>0.22</td>
</tr>
<tr>
<td>SRPH-3</td>
<td>80</td>
<td>Allen</td>
<td>Boron- No iron</td>
<td>8.30</td>
<td>8.33</td>
<td>0.33</td>
</tr>
<tr>
<td>SRPH-4</td>
<td>80</td>
<td>Allen</td>
<td>Boron- No iron</td>
<td>8.14</td>
<td>8.15</td>
<td>0.01</td>
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<td>SRPH-5</td>
<td>80</td>
<td>Allen</td>
<td>Boron- No iron</td>
<td>8.27</td>
<td>8.33</td>
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<td>SRPH-6</td>
<td>80</td>
<td>Allen</td>
<td>Boron- No iron</td>
<td>8.21</td>
<td>8.23</td>
<td>0.18</td>
</tr>
<tr>
<td>SRPH-7</td>
<td>80</td>
<td>Allen</td>
<td>Boron- No iron</td>
<td>8.31</td>
<td>8.33</td>
<td>0.23</td>
</tr>
<tr>
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<td>Allen</td>
<td>Boron- No iron</td>
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<td>8.49</td>
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<td>Allen</td>
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<td>8.74</td>
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<tr>
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<td>Allen</td>
<td>Boron- No iron</td>
<td>8.51</td>
<td>8.52</td>
<td>0.09</td>
</tr>
</tbody>
</table>

Note: SRPH= Hydrated UO₂ powder from Savannah River
Table III. Sintering of Attrition Milled UO$_x$ Powder  
(No Binder Added)

<table>
<thead>
<tr>
<th>Sample #</th>
<th>UO$_x$ content (vol. %)</th>
<th>Type of Clay Additive</th>
<th>Description of Mill Additions</th>
<th>Bulk Density (g/cc)</th>
<th>Apparent Density (g/cc)</th>
<th>Open Porosity %</th>
</tr>
</thead>
<tbody>
<tr>
<td>SRPH-11</td>
<td>80</td>
<td>Allen</td>
<td>Boron- No iron</td>
<td>8.47</td>
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<tr>
<td>SRPH-12</td>
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<td>Allen</td>
<td>Boron- No iron</td>
<td>8.84</td>
<td>8.79</td>
<td>0.00</td>
</tr>
<tr>
<td>SRPH-13</td>
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<td>Allen</td>
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<td>8.64</td>
<td>8.61</td>
<td>0.00</td>
</tr>
<tr>
<td>SRPH-14</td>
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<td>Allen</td>
<td>Boron- No iron</td>
<td>8.65</td>
<td>8.69</td>
<td>0.46</td>
</tr>
<tr>
<td>SRPH-15</td>
<td>80</td>
<td>Allen</td>
<td>Boron- No iron</td>
<td>8.80</td>
<td>8.78</td>
<td>0.28</td>
</tr>
</tbody>
</table>

Note: SRPH= Hydrated UO$_3$ powder from Savannah River
Figure 1. Micrographs of Hand-ground/mixed & Sintered DUAGG Sample
Figure 2. Micrographs of Attrition-milled/mixed & Sintered DUAGG Sample
(Binder Added by Hand)
Figure 3. Micrographs of Attrition-milled/mixed & Sintered DUAGG Sample
(No Binder Added)
References


7. BR-.100 Shipping Cask Preliminary Design Report, 51-1177082-03, Babcock & Wilcox Fuel Co., Page II-3-1 1.

8. American Concrete Institute Standard, ACI-349-30 Appendix A.4 thru A.4.3.
DUAGG/DUCRETE Development Requirements

DUAGG\textsuperscript{tm}/DUCRETE\textsuperscript{tm} Experimental Program

July, 1997

PREPARED BY:

LOCKHEED MARTIN IDAHO TECHNOLOGIES COMPANY

and

NUCLEAR METALS, INC.
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Table 1. Depleted Uranium Inventories in the DOE Complex .......... 1
DUAGG™/DUCRETE™ Experimental Program Future Proposed Work

Executive Summary

This paper outlines the sequence of events and associated work required to advance the DUAGG™/DUCRETE™ program into the production of licensed, spent nuclear fuel casks. The principle of forming an artificial aggregate using various compounds of uranium oxide (UO₂, UO₃, and U₃O₈) undergoing a liquid phase sintering process have been successfully demonstrated at the Idaho National Engineering and Environmental Laboratory (INEEL). Future work needs to include: (1) additional material properties testing in support of cask design data requirements, (2) depleted uranium concrete (DUCRETE) manufacturing studies, (3) cask design and review, (4) cask construction and testing, and (5) Safety Analysis Report (SAR) in support of storage license and Certificate of Compliance.

The most important work needing to be performed, in the near term, is DUAGG and DUCRETE materials property testing and manufacturing studies. These two activities can be performed concurrently and together provide the most critical need for cask design, construction, and testing.
1. Introduction

Depleted uranium inventories are found throughout the DOE Complex as well as some DOD facilities. Some private supplies of DU also exist but to a much smaller extent. Most (95%) of the DOE inventory exists in the form of uranium hexafluoride - UF₆. Other smaller but still significant quantities (>50 million pounds) of depleted uranium exists at DOE sites in other chemical forms. Table 1 below identifies the major known quantities and locations.

Very little of this depleted uranium material has a known future use.

This overall DU inventory situation requires integrated management or large and valuable material inventories may be wasted. If declared a waste, large liabilities may be incurred.

The Idaho National Engineering and Environmental Laboratory (INEEL) in partnership with private industry (Nuclear Metals, Inc.) is developing a method to produce high-density aggregate (artificial rock) primarily consisting of depleted uranium oxide suitable for use as a component of high-density concrete used as radioactive shielding.

<table>
<thead>
<tr>
<th>Location</th>
<th>Chemical Form</th>
<th>Quantity (MT)</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>Paducah</td>
<td>UF₆</td>
<td>*</td>
<td>28,351 full cylinders (10 and 14 tons each)</td>
</tr>
<tr>
<td>Portsmouth</td>
<td>UF₆</td>
<td>*</td>
<td>13,388 full cylinders (10 and 14 tons each)</td>
</tr>
<tr>
<td>Oak Ridge</td>
<td>UF₆</td>
<td>*</td>
<td>4,682 full cylinders (10 and 14 tons each)</td>
</tr>
<tr>
<td>Fernald</td>
<td>Umetal</td>
<td>2000</td>
<td>Planned for disposal in FY1998</td>
</tr>
<tr>
<td>Nevada Test Site</td>
<td>Umetal</td>
<td>70</td>
<td>Sheet metal scrap, Planned for disposal by September 1997</td>
</tr>
<tr>
<td>Savannah River</td>
<td>Umetal</td>
<td>950 1600</td>
<td>Bare metal Nickel plated-aluminum clad</td>
</tr>
<tr>
<td>Fernald</td>
<td>UF₄</td>
<td>2000</td>
<td>Planned for disposal in FY1998</td>
</tr>
<tr>
<td>Savannah River</td>
<td>UO₃</td>
<td>19420</td>
<td>Indefinite Storage</td>
</tr>
<tr>
<td>Fernald</td>
<td>UO₃</td>
<td>40</td>
<td>Planned for disposal in FY1998</td>
</tr>
</tbody>
</table>

* Total inventory at these sites is generally estimated to be 550,000 MT as of the transfer of the facilities to the US Enrichment Corporation in July 1993. Upon privatization, material generated since transfer (~2000 cylinders per year will also revert to DOE).
The oxide materials identified in Table 1 are sufficient to fabricate large quantities of depleted uranium concrete (DUCRETE™) containers for LLW, HLW or spent fuel storage over packs. The Savannah River UO₃ inventory is sufficient to fabricate over 500 spent fuel storage over packs. The shielded container application will require additional testing, beyond developmental testing, for validation of mechanical, chemical, and nuclear properties of both the DUAGG and depleted uranium concrete (DUCRETE™) materials. Additionally, manufacturing processes need to be optimized to reduce depleted uranium aggregate production costs.

2.0 Technical Needs

This document describes activities related to the maturation of DUCRETE technology which need be performed. Also included is the estimated time needed to resolve the issues under investigation.

2.1 Aggregate Production

For depleted uranium recycle to be cost effective, the process cost for converting UOₓ into aggregate (DUAGG) needs to be inexpensive and environmentally acceptable.

The conversion will be performed in two steps:

1.) Chemical conversion of the UF₆ to Oxide, and

2.) Production of aggregate from the oxide.

Processes have been identified for these conversion steps, however, process optimization is needed to verify that performance is maximized and costs are minimized.

2.1.1 Aggregate Production Optimization Studies

The INEEL has developed the bench scale process techniques for making DUAGG depleted uranium aggregate. Production of aggregate in any scale beyond a few kilograms has not yet occurred. Nuclear Metals, Inc. (NMI) has obtained a license for production of DUAGG from Lockheed Martin Idaho Technologies Company (LMITCO). NMI is installing pilot scale equipment for DUAGG production in its Carolina Metals facility located near Barnwell, SC. NMI has contracted with LMITCO for production of 6 tons of DUAGG for use in subsequent test programs. Initial production will follow the process steps developed at INEEL. Future work should involve experimenting with alternative processing steps to simplify processes and lower total production costs.

The current process has potential for cost reduction through simplification. For example dry versus wet grinding will save considerable energy and reduce the number of processing steps. Wet grinding was chosen to simplify contamination control. If dust control can be accomplished in the grinding operation, reduced operating costs can be assured. Alternative processes for reduction of UO₃ to UO₂ should also be investigated to reduce the sintering cycle time.

2.1.2 Economic Analysis

Most of the current cost estimates are based on simple conceptual designs of a
processing facility or upon processing concepts which now are nearly two years old. For establishing the incentive for process optimization and for developing the absolute cost for DUAGG production, a detailed facility design should be developed and life cycle costs estimated. Such a facility should be designed at a couple of capacity throughputs to meet the potential demand of different recycle scenarios.

2.2 DUCRETE Product Development

DUCRETE Concrete is proposed as a solution to an excess material problem. DUCRETE can be used as nuclear shielding but classic demand pull market forces will not cause customers to demand DUCRETE Shielding for their shielding needs. Most of the time, other traditional materials will work just fine and both costs and engineering parameters are better understood. Thus, nuclear engineers, being risk adverse, will stay with known materials. Consequently, to help the user community view DUCRETE as a viable solution to their shielding needs, DOE must create products and demonstrate their effectiveness. This section discussed data manufacturing needs and product demonstrations which will provide the technical basis for wide scale use of DUCRETE Concrete.

2.2.1 DUCRETE Manufacturing Technologies

No DUCRETE Concrete has been fabricated other than the initial work done as a proof of concept in 1994. This work used surplus depleted uranium oxide fabricated as reactor fuel pellets. These fuel pellets were far from an ideal aggregate due to both their size, shape and surface finish. Nevertheless, based on the results of compression tests, the concrete samples made from them clearly showed that DUCRETE concrete was feasible. However, since those initial tests, no additional DUCRETE concrete has been made.

The high density DUAGG aggregate is the basis behind the enhanced performance attributes of DUCRETE shielding. Most traditional concrete weighs about 140 lb/ft³. Theoretical estimates of DUCRETE concrete are near 450 lb/ft³.

At this density, several new issues need to be addressed:

1) Mixing methods,
2) Fabrication techniques,
3) Mixture stiffness in the wet state,
4) Optimization of super-plasticizer usage,
5) Control over aggregate settling, and
6) Aggregate size range in the mixture.

Methods to pour large concrete structures need to be tested and process steps developed. Mixing processes versus using pre-placed concrete methods need to be evaluated. Aggregate size fractions must be established for prevention of separation. Aggregate mixture densities need to be analyzed to assure homogenous distribution of aggregate. Samples need to be made, tested, and process modification identified as required.

2.2.2 DUCRETE Concrete Material Properties for Structural Applications

Once concrete mixtures have been reasonably established, additional mechanical, thermal, and nuclear property testing, supplementing development testing already performed, must be conducted. The experimental data needs to be acquired
under an approved Quality Assurance Program to establish the needed pedigree for subsequent use in licensing calculations and safety analysis reports. Test data should be acquired for two aggregate loading ratios (50% and 58% volume loading ratio). These additional or supplemental DUCRETE tests include:

**Mechanical Properties**

1) Unconfined compressive strength and tensile strength,

2) Ultimate tensile strength,

3) Density,

4) Bulk modulus (as a function of confining pressure),

5) Stress-strain constitutive relations,

6) Young's modulus, and

7) Poisson's ratio.

**Thermal Properties**

1) Thermal emissivity,

2) Thermal conductivity (0°C to 200°C),

3) Heat capacity, and

4) Coefficient of thermal expansion.

Additional nuclear properties measurements are necessary to validate development testing data and calculated gamma and neutron attenuation characteristics.

Conservative values will be needed for the design of any licensed spent fuel containers. These properties must be obtained in a statistically defensible quantity so that subsequent product design can use the information and support license applications.

### 2.2.3 DUCRETE High Activity LLW Containers

One application for DUCRETE shielding is expected to be for high activity low level waste containers needed for the storage of nuclear materials at DOE sites. To stimulate the use of DUCRETE containers, prototypes must be fabricated and the advantages demonstrated. After successful deployment, potential customers will be more convinced of the material effectiveness. Two potential applications for such containers have been identified at Fernald (~2000 rectangular Type A boxes) and at INEEL (~675 cylindrical drum over packs). Together, these applications will consume 18 million pounds of uranium oxide. This task will develop detailed designs, fabricate prototype containers, test the containers, and develop detailed cost estimates for production quantities.

### 2.2.4 DUCRETE Spent Fuel and HLW Storage Casks

The largest potential application for DUCRETE shielding involves the storage of spent nuclear fuel for DOE Test Reactor Fuel, US Navy Submarine Reactor Fuel, Commercial Power Reactor Fuel, and High Level Waste from the DOE Vitrification facilities. Efforts need to be undertaken to develop (design and manufacture) and test a full scale reactor HLW or spent fuel storage cask.

### 2.3 UF₆ Conversion to Stable Materials

The UF₆ stored at the gaseous diffusion plants was an excellent material for the cascade operations but represents a very poor compound from a long term management perspective. UF₆ is a solid at storage
temperatures but sublimes at slightly elevated temperatures. Upon contact with water, it react to form HF gas and UO$_2$F$_2$, a water soluble solid. The carbon steel cylinders that hold the UF$_6$ have heavily oxidized due to outdoor storage. Consequently, some conversion process will ultimately be needed to convert UF$_6$ to a more stable final form such as uranium oxide (UO$_3$, U$_3$O$_8$ or UO$_2$). Conversion technologies haven't changed much over the last 40 years. Virtually all proven conversion processes involve some elevated temperature reaction of UF$_6$ with steam. Most reactions produce uranium oxide and aqueous hydrogen fluoride. Ideally, the attributes of the conversion process will minimize the cost, minimize the waste generation, and maximize the value of any product residuals. Even though research needs to be conducted in this area this white paper will not address any conversion processes currently in use or proposed.

3.0 Summary

It is estimated that completion of the additional material testing can be completed in two years with cask design and cask testing completed in four years. This effort will develop the technical basis for the economical production of DUAGG, establish its chemical stability, develop DUCRETE technology, and demonstrate the technical performance of products fabricated from such materials. This technical basis will provide the necessary knowledge and the experience using this new material for deployment and licensing of nuclear shielding systems.
White Paper

DEPLETED URANIUM OXIDES AS SPENT-NUCLEAR-FUEL WASTE-PACKAGE FILL MATERIALS

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Manuscript Date: July 7, 1997

Prepared for

U.S. Department of Energy Depleted Uranium Workshop
Las Vegas, Nevada
July 15, 1997

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Managed by Lockheed Martin Energy Research Corp. under contract DE-AC05-96OR22464 for the U.S. Department of Energy.
DEPLETED URANIUM OXIDES AS SPENT-NUCLEAR- FUEL
WASTE-PACKAGE FILL MATERIALS

Charles W. Forsberg

ABSTRACT

A new technology has been proposed to use depleted uranium dioxide (UO₂) as a fill material in spent nuclear fuel (SNF) waste packages (WPs). Empty WPs at the geologic repository are loaded with SNF. The void space between the fuel pins and the outer void space between SNF assemblies and the inner WP wall are then filled with small depleted UO₂ particles. The repository WPs are then sealed and placed into the repository.

This concept provides: (1) shielding, (2) reduced potential for repository nuclear criticality events, and (3) reduced long-term release of radionuclides from the WP in the repository. The presence of depleted uranium (DU) fill reduces the radiation exposure to repository workers and the local geology while reducing the external WP dimensions. The concept minimizes short-term and long-term nuclear criticality concerns in the repository by reducing the enrichment of the WP to well below 1 wt % ²³⁵U equivalent. Finally, the concept reduces the long-term release of fission products and actinides from the WP into the environment. Most of these radionuclides are incorporated within the SNF UO₂ pellets of the SNF. These radionuclides can not be released until the WP fails and the SNF UO₂ crystal structure is destroyed, thus allowing release of trapped radionuclides. The depleted UO₂ preferentially reacts with groundwater and suppresses the dissolution of the SNF UO₂ after the WP has failed. It does this by multiple chemical mechanisms: maintenance of chemically reducing conditions within the WP, saturation of the groundwater in the degraded WP with DU, and reduction of degraded WP permeability to air and water flow through the WP. These beneficial properties exist only for DU fills because only the depleted UO₂ has the same chemical form as the light-water reactor SNF uranium.

The use of DU as a fill material is independent of the use of DU in the external WP or as a backfill material around and outside of the WP. The concept is independent of the cask concept chosen, although it may impact that cask design (e.g., reduce required wall thickness). The fill could also be added to transport and storage casks when the inner container is to be sent to the geological repository for disposal.
As a new concept, there are significant uncertainties. The primary issues to be resolved are fill physical and chemical properties (particle size and size distribution, thermal conductivity, etc.), impacts on WP design, and post-closure performance.

1. INTRODUCTION

The use of depleted uranium (DU) as a repository waste-package (WP) fill material is being investigated\(^1\). A synthesis of existing laboratory data on uranium oxides, hot-cell tests on spent nuclear fuel (SNF), geological field data, and other data provide the technical basis for the concept. A description of the concept is provided. The mechanisms are described for (1) reduction of long-term radionuclide release rates from the WP, (2) minimization of the potential of long-term nuclear criticality in the repository, and (3) reduction of the radiation levels near the WP. The major uncertainties are defined, and the required development activities are discussed.

2. CONCEPT: APPLICATION TO A REPOSITORY

The repository WP is loaded with SNF. The WP void space is then filled with depleted uranium dioxide (UO\(_2\)) particulates, which are sufficiently small (<1 mm) such as to fill coolant channels in the SNF assemblies and spaces between the fuel assemblies and the interior wall of the WP. A simplified schematic of a loading method is shown in Fig. 1. Table 1 identifies key characteristics of a representative WP using DU fill. The DU particulate enrichment levels are as low as 0.2 wt % U. Thus, DU added to the package reduces the total fissile material concentration in the WP to below 1 wt % of the heavy metal. In the base case, the DU fill would be loaded at the repository. The technology is applicable to light-water reactor (LWR) SNF and potentially applicable to high-enriched-uranium SNF.

It is expected that the use of fill material will have little impact on the design of the external WP. The external WP walls can include added DU, if desired. The WP would probably be filled with the DU UO\(_2\) fill at the repository, but the fill could also be added to transport and storage casks when the inner container is to be sent to the geological repository for disposal.
Fig. 1. WP system and example loading sequence for DU particulates.
Table 1. Depleted uranium dioxide WP fill characteristics

<table>
<thead>
<tr>
<th>Property</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>SNF (metric tons initial heavy metal)</td>
<td>9.96</td>
</tr>
<tr>
<td>Solid bead density (g/cm²)</td>
<td>10.96</td>
</tr>
<tr>
<td>DU (wt % of fill)</td>
<td>88</td>
</tr>
<tr>
<td>DU mass (t)b</td>
<td>32.3</td>
</tr>
<tr>
<td>Particulate mass (t)b</td>
<td>36.7</td>
</tr>
<tr>
<td>Assay (wt %) of DU $^{235}\text{U}$</td>
<td>0.2</td>
</tr>
<tr>
<td>Equivalent $^{235}\text{U}$ assay (wt %) of SNF</td>
<td>1.6</td>
</tr>
<tr>
<td>Equivalent $^{235}\text{U}$ assay (wt %) of WP</td>
<td>0.53</td>
</tr>
</tbody>
</table>

The design basis was the U.S. Multi-Purpose Canister (MPC) WP system for 21 pressurized-water reactor SNF assemblies. The MPC has an internal volume of 7.9 m³, the SNF baskets have a solid volume of 1.1 m³, and the 21 SNF assemblies have a solid volume of 1.6 m³. A total of 5.2 m³ of the 7.9 m³ of internal volume can be filled with beads.

The fill fraction is 0.64; i.e., 64% of the initial void space is filled with UO₂ particulates and 36% of the initial void space remains void space between the particulates. Large-scale Canadian tests have measured volume fill fractions up to ~73%. Manufacturing techniques can make, if desired, UO₂ with much lower densities.

3. REPOSITORY BENEFITS

3.1 REDUCTION OF RADIONUCLIDE RELEASE RATE FROM THE REPOSITORY WP

The expected repository failure mode is radionuclide migration to the open environment by (1) WP failure, (2) leaching of SNF by water, (3) dissolution of radionuclides and generation of colloids, and (4) transport of those radionuclides in dissolved or colloidal forms to the open environment. Two mechanisms reduce radionuclide release rates: (1) minimizing the water flow through the WP and (2) reducing the dissolution of radionuclides into groundwater. The use of DU fill slows each of these radionuclide release mechanisms by multiple phenomena for geological repositories with oxidizing conditions, such as the proposed repository at Yucca Mountain.
3.1.1 Maintenance of Chemically Reducing Conditions to Minimize SNF UO₂ Failure

Most fission products and actinides within SNF are incorporated within UO₂ crystals in SNF pellets. Upon WP failure and after entry of groundwater into the WP, these fission products and actinides cannot escape the UO₂ crystals until the UO₂ dissolves or is transformed into another chemical species.

Maintaining chemically reducing conditions within the WP maintains the integrity of the SNF UO₂ crystals by multiple mechanisms; and hence, reduces radionuclide releases. With the use of UO₂ or other uranium compounds as fill materials with uranium in the +4 valence state, the DU fill ensures chemically reducing conditions within the WP for an extended time independent of external groundwater chemistry. After WP failure, any oxygen in the groundwater would first encounter the DU before it encounters the SNF. The DU in the +4 chemical state would be oxidized to the +6 chemical state through a series of oxidation steps, thus removing the oxygen from the groundwater and maintaining chemically reducing conditions in the WP.

The solubility of uranium in reducing groundwater is very low—about 1 ppb. This solubility is up to four orders of magnitude less than the solubility of uranium under oxidizing conditions. Radionuclide releases from SNF are extremely limited under chemically reducing conditions because the UO₂ does not dissolve and release the radionuclides incorporated within its crystalline structure. Chemically reducing conditions also minimize the solubility and transport of several other long-lived radionuclides (e.g., neptunium and technetium) and reduce the formation and transport of radionuclides as colloids.

Chemically reducing conditions also prevent the conversion of SNF UO₂ to higher uranium oxides such as U₅O₈. When UO₂ is oxidized, its crystal structure changes. In the process of converting from one crystal form to another, some radionuclides are released from the matrix. In particular, noble gases and some soluble radionuclides (if water is present) will be released to the air and water in the WP.

The DU fill minimizes generation of oxidizing acids that degrade the WP and accelerate SNF dissolution. Water and air in radiation fields generate corrosive acids (e.g., nitric acid). WPs are dried and filled with inert gases before sealing to minimize this problem. After WP failure, DU fill reduces acid generation by two mechanisms. The high-density fill reduces internal WP radiation fields. The available WP air or water volume for acid generation is reduced by displacement with DU fill. Each of these mechanisms reduces acid generation proportionally.
3.1.2 Saturation of Groundwater with DU

Surrounding the SNF with DU saturates any groundwater which may enter a degraded WP with uranium. This saturation slows the SNF dissolution process. DU-saturated groundwater can not dissolve additional SNF uranium.

3.1.3 Support of WP Integrity

Fill material helps maintain other barriers to radionuclide migration. Ultimately, the WP will lose its structural strength. With a fill material, the basic geometry of the WP is maintained after loss of structural integrity of the WP. Without a fill material, the WP will (1) itself collapse, (2) collapse any secondary barriers to water flow, and (3) create a more permeable bed of rubble to groundwater flow. This benefit applies to any fill material, not just to DU.

3.1.4 Protection from Variable Groundwater Chemistry

The use of DU, particularly UO$_2$, provides a mechanism to counteract any unexpected changes in groundwater that may accelerate degradation of the SNF. Groundwater chemistry can change because of climatic changes and human activities (e.g., irrigation, groundwater pumping, liquid-waste injection, etc). The DU UO$_2$ acts as a sacrificial chemical to absorb changes in groundwater chemistry and delay their effects on the SNF UO$_2$.

3.1.5 Minimization of the Groundwater Flow in WP

After WP failure, DU fill lowers the hydraulic conductivity within the degraded WP and, thus, ensures low groundwater flow near the SNF. Water flow is the primary mechanism for the transport of radionuclides to the open environment. The release of many radionuclides is controlled by solubility limits. Other radionuclides may be transported by colloids. The slower the flow of groundwater, the slower the transport of radionuclides. The goal is to create a WP where water flow and radionuclide migration are by diffusion only because diffusion is a very slow process.

In oxidizing groundwater, the DU oxide fill will evolve to lower-density, higher-valence state, hydrated, uranium oxides$^{4-6}$. Over much longer time periods these compounds will evolve further into hydrated uranium silicates. The silica is provided by the groundwater. These changes have been observed in the laboratory and in uranium ore deposits$^{7-8}$. This reduction in density and accompanying volume expansion fills the void space and, thus, minimizes water flow. Figure 2 shows this evolutionary sequence as it has been observed in natural uranium ore bodies with oxidizing groundwater$^9$. Because of the surface area of the particulates and because they are not protected by the SNF clad, they will expand and choke off groundwater flow to the SNF. Figure 3 shows the conversion of UO$_2$ to U$_3$O$_8$ that initially occurs. The desired expansion can be controlled by design.
Initial Conditions

Conversion of $\text{UO}_2$ to $\text{U}_3\text{O}_8$

Fig. 3. Conversion of $\text{UO}_2$ to $\text{U}_3\text{O}_8$ with reduction in WP water and air permeability.
3.2 REDUCTION OF THE POTENTIAL FOR REPOSITORY NUCLEAR CRITICALITY

Both short-term and long-term nuclear criticality are to be avoided in a geological repository. A nuclear criticality event would generate added radioactivity and heat. The heat can accelerate degradation of WPs and movement of water that may transport radionuclides to the environment. The added radioactivity and heat also creates uncertainties in the modeling of the long-term performance of the repository. These and other considerations have lead to the licensing requirement that nuclear criticality be avoided in a geological repository.

The use of DU as a fill material reduces the potential for repository nuclear criticality events by lowering the fissile assay of a WP below 1 wt % $^{235}$U. The average enrichment of SNF (all fissile isotopes) is $\sim$1.5 wt %. There is a wide distribution of fissile concentrations within the SNF inventory. It is generally accepted that a nuclear criticality event will not occur in geologic time at enrichments <1 wt % $^{235}$U equivalent. Criticality is prevented in a repository by (1) geometric spacing of fissile materials and (2) neutron absorbers. The degradation of the WP and SNF will ultimately change the geometry of the WP and, thus, complicate the use of geometry control for long-term criticality control. Neutron absorbers include $^{238}$U, boron, gadolinium, and other materials. Because of possible uranium dissolution, groundwater transport, and redeposition (a mechanism that creates uranium ore bodies), it has been suggested that the potential for criticality events exists if the fissile concentration in the repository SNF is sufficiently high. Neutron absorbers (except $^{238}$U) leach from WPs and travel at rates different from the SNF uranium through the geologic media because of the different chemistries of the neutron absorbers in groundwater.

The separation and concentration mechanisms for uranium in a repository are the same as those for uranium in the natural environment. In effect, the same phenomena that created natural reactors in the distant past may create the potential for nuclear criticality events in the future. Uranium under oxidizing conditions is several orders of magnitude more soluble than uranium under chemically reducing conditions. This allows uranium to be oxidized to the +6 valence state by oxidizing groundwater, dissolve in the groundwater, be transported by the groundwater, and precipitate from groundwater when the local geological conditions create chemically reducing conditions. Potential chemical reducing agents are natural organics and many WP materials, such as iron. Figure 4 shows one such scenario. The use of DU as a neutron absorber minimizes these concerns because DU has the same chemical properties as SNF uranium and, thus, does not separate from the SNF uranium. It minimizes potential criticality concerns inside, near, and far from the WP.
FORMATION OF URANIUM ORE DEPOSITS FROM URANIUM IN ROCK

RAIN

Dissolve uranium in oxidizing groundwater

Roll front uranium deposit
\[ \text{U}^{6+} \rightarrow \text{U}^{4+} \]

Oxidizing groundwater

Reducing groundwater (little uranium)

Reducing geology (organic, etc.)

FORMATION OF URANIUM ORE DEPOSITS FROM URANIUM WASTES

RAIN

Oxidizing groundwater

Dissolved uranium in groundwater

Degraded waste package

Iron (waste package, rock bolts, etc.)

Reducing groundwater (little uranium)

Fig. 4. Natural and man-made formation of uranium ore bodies.
In LWR SNF, much of the fissile material is $^{239}$Pu. The foregoing analysis is based on the assumption that plutonium remains with the uranium until the major plutonium isotope ($^{239}$Pu) decays to $^{235}$U and can be isotopically diluted by the DU. This isotopic dilution is ensured if the rate of plutonium decay to uranium is faster than the rate of dissolution and transport of uranium within the repository. Plutonium-239 has a half-life of 24,000 years (i.e., the decay rate is $3 \times 10^{-5}$/year). Performance assessments indicate that plutonium migration is slow in most geological environments; thus, DU fill provides a basis for long-term criticality control of $^{239}$Pu and its decay product $^{235}$U.

DU can be added exterior to the WP for criticality control; however, this has additional uncertainties compared to the use of a DU fill material. It eliminates the chemical benefits of DU within the WP. Furthermore, it adds uncertainties in terms of nuclear criticality control. Water can channel through a WP without fill material, transport the SNF uranium, and bypass DU exterior to the WP. Alternatively, the DU in the external package can be preferentially leached from the WP. These uncertainties increase with improvements in WP performance that allow for more evolution of the geological conditions outside the WP before WP failure. Last, DU exterior to the WP does not address internal-criticality issues associated with large WPs.

3.3 ENHANCED SHIELDING

WP shielding is required to reduce radiation interactions between the WP and the local geologic environment. This minimizes the complications in assessing long-term WP and repository performance. WP shielding may also be used to minimize the need for shielded equipment to place WPs in the repository. The use of a DU fill reduces radiation levels to the WP. This allows a reduction in shielding to be incorporated into the WP or lower radiation levels external to the WP. This shielding effect of the DU fill decreases the physical size of the WP because it is incorporated into the unused void spaces inside the WP.

4. SCIENTIFIC BASIS OF DESIGN

A focus of repository design is on the prediction of long-term performance over geologic time. Because the kinetic dissolution mechanisms of WP and SNF degradation may be temperature dependent and influenced by radiolysis rates, results from accelerated laboratory testing of materials are difficult to extrapolate to repository timeframes. Knowledge of uranium ore body behavior provides useful insight. In ore deposits, UO$_2$ evolves into hydrated uranium silicate minerals in conditions similar to those at Yucca Mountain. In these ore deposits, uranium at the outer edges of the deposits protects internal uranium against dissolution. The use of DU fill material in SNF WP mimics this natural phenomenon. Geological data on the performance of UO$_2$ over millions of years can be used, when combined with laboratory data, to provide estimates of WP performance and support licensing.
Many natural uranium deposits contain sufficient \( ^{235}U \) mass to become nuclear reactors. However, this does not happen because \( ^{235}U \) in natural uranium is isotopically diluted with \( ^{238}U \). The DU fill strategy applies this same strategy for criticality control to WPs.

5. STATUS

Initial studies have defined the concept. Key input data are primarily from the U.S., Canadian, and European repository programs which have performed extensive research on the degradation of SNF, the evolution of uranium ore deposits, and the modeling of WPs. There exists a large data base of theoretical and experimental information, but the information was collected in different programs for different applications under different repository conditions.

5.1 FILL TESTS

Limited tests of filling dummy LWR fuel assemblies with steel shot have been done\(^ {15} \). Canada\(^ {16} \) has conducted extensive tests to fill empty spaces with small particles within WPs containing simulated CANadian Deuterium Uranium (CANDU) SNF. The Canadian WP is a thin-walled, titanium container in which the SNF and fill material must support the WP against external hydrostatic pressure after burial. Canadian tests investigated different particle sizes, different mixtures of particle sizes, alternative fill materials, vibratory filling with different vibration frequencies and amplitudes, and other factors. Tests included successful full-scale WP hydrostatic tests (10 MPa, 150°C) of the filled WP. These large-scale experiments indicate that the maximum practical packing density is about 70%\(^ {17-18} \), i.e., in a well-packed container, the solid particulate volume is about 70% of the initial void space and 30% of the initial void space remains as voids.

A representative design of WP with DU fill is shown in Table 1. The \( \text{UO}_2 \) fill fraction is assumed to be that demonstrated in Canadian fill tests (using other fill materials). The DU fill has a smear density (average over \( \text{UO}_2 \) particulates and void space) of \(-7 \, \text{g/cm}^3 \). It is noted that the solid density of manufactured \( \text{UO}_2 \) particulates can be varied widely and, thus, lower smear fill densities can be chosen.

5.2 DU CHEMICAL FORM

The baseline fill material is \( \text{UO}_2 \). There are alternative uranium compounds that can be used\(^ {15} \). These include other uranium oxides, uranium silicates, and uranium glasses. In the natural environment, \( \text{UO}_2 \) under
oxidizing conditions evolves to higher-valence uranium oxides that eventually become hydrated uranium silicates (see Fig. 2). It is desirable that the fill material have a composition similar to uranium compounds found in this chronological sequence to minimize the uncertainties in the long-term behavior of uranium in the WP. Uranium dioxide was chosen as the base line concept for several reasons: (1) it is fully compatible with LWR SNF that is composed of U0₂ in Zircalloy clad, (2) it is a well understood uranium compound, (3) its manufacturing techniques are well understood, and (4) it maximizes the use of DU (highest density).

5.3 THERMAL ANALYSIS

A thermal analysis was completed of a WP using the properties of sand as a fill material. Somewhat similar results are expected for U0₂. The use of sand as fill material had only limited effects on WP temperature. This was a direct result of the design of LWR SNF and the WP interior. The spent fuel clad and the internal WP grid structure are built of high-thermal conductivity metals. As a consequence, the heat flow path from the interior to exterior is (1) heat in the center pin in the center fuel assembly flows from the fuel to the clad, (2) heat flows across the interface to the next fuel pin, (3) heat is conducted through the clad to the opposite side of the pin, (4) heat is transferred from pin to pin until it reaches the WP grid structure, (5) the heat is conducted to the main body of the cask via the grid structure, and (6) the heat flows through the cask wall. The heat flow path is shown in Fig. 5. In typical designs, the heat will conduct through a few centimeters of fill; thus, the sensitivity of the design to the thermal conductivity is reduced.

5.4 SHIELDING ANALYSIS

The addition of fill material reduces radiation levels from the SNF to the WP wall. If the WP design objective is a partly or fully self-shielded WP, the use of DU fill allows a reduction in the thickness of the WP wall and, thus, a reduction of the external WP dimensions. This, in turn, may allow a reduction of repository tunnel size. Earlier analysis, with uranium glass fills (smear fill density ~2.6 g/cm³) indicated that the addition of fill material did not significantly change the total weight for self-shielded WPs. The fill adds weight but reduces the WP wall thickness and, hence, weight needed for radiation shielding. Detailed calculations have not been performed for U0₂, but a somewhat similar result is expected.

5.5 SCIENTIFIC ACCEPTANCE

Any new concept will be reviewed by the scientific community and must be ultimately accepted by the scientific community. The U.S. Nuclear Waste Technical Review Board (NWTRB) was created by the U.S. Congress to provide technical review of the Yucca Mountain Project. In the most recent evaluation of the project, the NWTRB observed that:
Fig. 5. Major heat-transfer paths in the MFC and other WPs.
"The potential advantages of fillers include providing additional structural support inside the WP, reacting with and sequestering certain radionuclides, providing additional shielding, and retarding the flow of water to and from the waste form. The added structural strength would reduce the size of water-collecting depressions over the spent fuel after the package is crushed."

These observations led to the recommendation:

"Given the inevitable uncertainties about repository performance, more attention to defense-in-depth (multiple, redundant barriers) is needed in the WP and repository designs. In particular, comprehensive studies of alternative engineered barriers—such as fillers, backfill materials, drip shields, and engineering inverts—should be completed."

The NWTRB also reviewed strategies to ensure that nuclear criticality is not a major issue in the licensing of the repository and made the following recommendation:

"In particular, the use of depleted uranium in filler, invert, or backfill material, or in all three, is a concept the program has not yet explored adequately. Conceivably, increasing the criticality control robustness of the EBS (Engineered Barrier System) could turn a potentially intractable analysis of external criticality into a comparatively easy one.

6. TECHNICAL ISSUES

The primary technical issues are: (1) the demonstration of a design, (2) licensing of the DU fill design in a geological repository, and (3) quantifying DU fill benefits as a function of the design parameters. The decision to use DU fill depends upon its impact on repository performance and cost factors. The issue is being simplified by the collection of experimental data from Yucca Mountain tunneling efforts that are providing a better definition of initial geological conditions.

Sufficient experiments have not yet been conducted to support the development of this concept. Most of the issues directly or indirectly depend upon the specific selection of the DU fill material. The particulate fill chemical and physical forms are the critical design parameters that impact all aspects of the design. Key design parameters include particulate size, particulate-size distribution, particulate density, chemical form, and chemical reactivity to air and water. Chemical form includes the choice of chemical species (baseline: UO$_2$) and the allowed range of impurities. Production techniques allow manufacture of DU oxide and other particulates with (1) different particle sizes, (2) different particle size distributions, (3) different solid densities, (4) different oxidation states, and (5) different chemical reactivities. Table 2 summarizes key technical issues.
<table>
<thead>
<tr>
<th>Physical and chemical properties of fill material</th>
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<tr>
<td>Particle size and size distribution</td>
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<td>Thermal conductivity</td>
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<td>Flowability</td>
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<td>Chemical reactivity (air and water)</td>
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<td>Permeability</td>
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<th>Waste package design</th>
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<th>Postclosure Performance</th>
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<tr>
<td>Establishing chemically reducing conditions</td>
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<tr>
<td>Swelling to reduce water and air permeability</td>
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<tr>
<td>Saturation of failed WP groundwater with DU</td>
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<td>Isotopic dilution of SNF uranium with DU</td>
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<th>Economics</th>
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<td>Technology transfer</td>
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<tr>
<th>Licensing</th>
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<tr>
<td>Concept demonstration</td>
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</table>
6.1 THERMAL AND MECHANICAL DESIGN

There are two mechanical design issues. First, WP fill tests have been conducted with particulates with densities lower than nominal UO₂. Validation of fill tests (laboratory and ultimately full-scale) with higher density materials is required, or the lower-density UO₂ must be used. Second, additional work is required to determine if there are design constraints in terms of peak SNF temperature in the WP when fill materials are used. The existence of a design constraint depends upon the total package design because peak SNF temperatures depend upon the entire flow path of heat from the center of the WP to the exterior. The fill material covers only a small distance between fuel pins and the outside WP wall. The overall thermal conductivity of various UO₂ fills varies as a function of particle size, density, and gas composition. Calculational methods can define the sensitivity of the peak temperature to estimated fill thermal-conductivity properties. If the fill thermal conductivity is a controlling design parameter, experimental measurements will be required.

6.2 REDUCTION OF LONG-TERM RADIONUCLIDE RELEASE RATE FROM THE WP

Modeling and experimental activities are required to determine the effects of DU fill materials on the radionuclide release rate. These activities must be integrated into current WP experimental and modeling activities. Because much of the repository SNF WP research program is investigating the behavior of SNF UO₂ in a WP to determine release rates of radionuclides from the WP, much of the ongoing activity is directly relevant to the use of DU fill materials. One clearly identified area of additional work is on models and experiments of the change of DU fill permeability, mechanical stresses, and other properties as air and water react with the fill material.

6.3 LONG-TERM NUCLEAR CRITICALITY CONTROL

The repository program is evaluating strategies long-term criticality control. The use of DU fill to control criticality is a different strategy than the current strategy. Significant investigations will be required in this area. These investigations primarily will be to refine models designed to predict the long-term chemical behavior of the WP. Criticality control is assured if isotopic exchange occurs between SNF UO₂ and DU fill during the degradation of the WP over time. In effect, with this strategy for criticality control, demonstrating that nuclear criticality will not occur depends upon understanding the chemical degradation of the WP over time.
6.4 ECONOMICS

No economic analysis has been done. The key question is what are the relative management costs of SNF and DU separately vs the use of DU in WPs. This is strongly dependent upon the requirements for disposal of SNF and the requirements for management of DU.

7. CONCLUSIONS

DU fill inside the WP creates the potential for significant improvements in WP performance based on uranium geochemistry, reduces the potential for nuclear criticality in a repository, and consumes DU inventory. As a new concept, significant uncertainties exist.

8. REFERENCES


5. W. M. Murphy, Technology Today (June 1992).


Background Information For

DOE Workshop on the Potential Uses of Depleted Uranium in the Repository

July 15 to 17, 1997

CONTENTS

Material Prepared By

Bill Quapp, Paul DeLozier, and Martin Haas

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3. SRS High Level Waste Canister Storage System Using DUCRETE Overpacks
4. Options For Using or Disposing of Depleted Uranium in the DOE Geologic Repository

INFORMATION PAPERS

1. Impact of the DUCRETE Cask on the Performance Assessment of the Repository
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11. Depleted Uranium Silicate Container Backfill System (DUSCOBS)

07/12/97
Introduction:

This Information Paper provides a brief description of the DUCRETE Cask conceptual design that is proposed as an overpack for the waste package being disposed in the DOE geologic repository. This overpack would be used first at the utility site for dry fuel storage until an Interim Storage Facility was available. It would be then transported for re-use at the Interim Storage Facility for dry fuel storage. After the repository is ready to receive fuel, the overpack would be transported a final time to the repository for assembly with the waste package to provide a self-shielded disposal package. The overpack is always transported empty. Fuel is transported separately in conventional, licensed transportation casks.

Cask Description:

DUCRETE concrete is the name given to a concrete formulation that uses depleted uranium aggregate instead of conventional aggregate resulting in a very high density concrete. Depleted uranium aggregate and depleted uranium concrete were conceived, developed and patented by researchers at the INEEL as a potential beneficial use of depleted uranium (Reference 1). Depleted uranium aggregate is manufactured from uranium oxide in a conventional ceramic manufacturing process. A more detailed presentation of DUCRETE concrete and its properties can be found in References (2 and 3). The DUCRETE cask concept used in this evaluation is very similar in design to other ventilated concrete storage casks produced by commercial cask vendors such as Sierra Nuclear, NAC and HOLTEC. It is briefly outlined in the following description. This concept is only one of several DUCRETE concrete options being explored as potential beneficial uses of depleted uranium. The design parameters of the DUCRETE concrete can be tailored to achieve a range of nuclear shielding parameters. The aggregate can also be tailored to have varying nuclear attenuation characteristics. A comparison of the shielding effectiveness between DUCRETE concrete and other shielding materials for gamma and neutron source terms for a spent fuel application is shown in Figure 1.

Figure 2 shows a schematic design of the DUCRETE ventilated storage cask used in this evaluation. Such a cask can accommodate approximately 21 PWR or 50 BWR spent fuel assemblies. A unique feature of this concept is that it provides sufficient space inside of the overpack for the waste package to be inserted just before emplacement into the mountain. Other traditional concrete overpacks developed by vendors such as Sierra Nuclear, including the DUCRETE cask design concepts evaluated in Reference 2, do not provide diametral clearance for the waste package.

The cask is a right circular cylinder about 17 feet long and 8 feet in diameter. The body of the cask is made of DUCRETE concrete with a density of approximately 400 lb/ft³. The cask walls are thick enough (10 to 12 inches) to provide sufficient gamma and neutron shielding to allow direct handling operations at a reactor site (about 25 to 50 mR/hr before insertion of the disposal package). The weight of the empty overpack is about 99 tons and the weight of the total disposal
package (overpack, fuel canister, fuel, and waste package) is 150 tons. The DUCRETE cask is completely encapsulated in 0.5 inches of stainless steel plate. The steel provides structural strength, protects the surfaces of the DUCRETE, provides a foundation for the attachment of other needed design features (such as lifting lugs) and provides a barrier for isolation of the DU from the environment.

For operation at reactor sites, inlet and outlet airflow ducts are located near the bottom and top of the cask to enable natural circulation cooling of the inner SNF package. The bottom intake ducts open into an annular empty space (nominally 2 inches) between the inner SNF package and the inside steel liner of the cask. Heat from the spent fuel inside the inner package is transferred to the air in this space; as its temperature increases, the air becomes less dense and flows up through the annular space and out the exhaust ducts. The dominant heat-transfer mechanism is natural convection while some heat is transferred by radiation to the cask inner wall. This reduces the amount of heat that flows through the DUCRETE and results in a lower temperature in the DUCRETE concrete.

Since DUCRETE is significantly more dense than the concrete used in similar concrete cask designs by other vendors the overall dimensions and wall thickness of the DUCRETE cask are corresponding smaller than these other casks. The DUCRETE cask is about 35 tons lighter, 40 inches smaller in diameter and length than conventional concrete storage casks currently available (Reference 2). The provision of additional space for the waste package increases the size and weight of the cask compared to that designed by Hopf (Reference 2).

Cask Transportability:

The relatively small diameter of the DUCRETE cask, even with the allowance for the addition of the waste package, allows factory fabrication and rail transport eliminating any requirements for field construction of the casks. Once licensed for general application, this innovation permits potential users to order new DUCRETE casks to meet their particular needs with a shorter manufacturing lead time and lower costs than field fabricated casks. Rail transportation of the DUCRETE cask also allows transport and reuse of casks that could potentially provide for optimizing their use in an integrated system approach to SNF storage and disposal. The transportation comparison between DUCRETE Casks and traditional concrete cask is illustrated in Figure 3.

Disposal Configurations:

Two disposal configurations have been considered and are illustrated in Figures 4 and 5. The concept shown in Figure 3 disposes the DUCRETE cask with waste package in a drift in a horizontal orientation. This configuration would require overpack development to achieve suitable natural convection cooling in this orientation. Normal cask orientation at reactor sites is vertical. In Figure 4, the DUCRETE cask with waste package is shown in a large drift in a vertical orientation. This imposes no development requirements on cask design but does impact repository design.
References:

1) U.S. Patent, Radiation Shielding Composition, W. J. Quapp and P. A. Lessing, Allowed but not yet issued.


Figure 1. Comparison of Wall Thickness to Attenuate Neutron and Gamma Doses to 10 mR/h From 24 PWR Spent Fuel Assemblies

J. Sterbentz, INEEL
Figure 2.

Schematic of Ducrete Overpack with Disposal Package.
Figure 3. Transportability Concept for DUCRETE Casks

Conventional Concrete Cask are Replicated at Each Location

Transportability of DUCRETE Cask Provides Multiple Advantages and Lower Overall Cost
Figure 4. Self-Shielded Disposal Overpacks
Using DUCRETE Concrete Overpacks In Drifts

Figure 5. Self-Shielded Disposal Overpacks
Using DUCRETE Concrete Overpacks In "Super-Drift"
DEPLETED URANIUM HEXAFLUORIDE:
THE SOURCE MATERIAL FOR ADVANCED SHIELDING SYSTEMS

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This paper was prepared for the
Third International
Uranium Hexafluoride Conference:
Processing, Handling, Packaging, Transporting

November 28 - December 1, 1995
Paducah, Kentucky

Meeting was sponsored by the
Institute of Nuclear Materials Management

The work was supported by the
Department of Energy
Environmental Management Office of Science and Technology

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DEPLETED URANIUM HEXAFLUORIDE:
THE SOURCE MATERIAL FOR ADVANCED SHIELDING SYSTEMS

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ABSTRACT

The U.S. Department of Energy (DOE) has a management challenge and financial liability problem in the form of 50,000 cylinders containing 555,000 metric tons of depleted uranium hexafluoride (UF₆) that are stored at the gaseous diffusion plants. DOE is evaluating several options for the disposition of this UF₆, including continued storage, disposal, and recycle into a product. Based on studies conducted to date, the most feasible recycle option for the depleted uranium is shielding in low-level waste, spent nuclear fuel, or vitrified high-level waste containers. Estimates for the cost of disposal, using existing technologies, range between $3.8 and $11.3 billion depending on factors such as the disposal site and the applicability of the Resource Conservation and Recovery Act (RCRA). Advanced technologies can reduce these costs, but UF₆ disposal still represents large future costs.

This paper describes an application for depleted uranium in which depleted uranium hexafluoride is converted into an oxide and then into a heavy aggregate. The heavy uranium aggregate is combined with conventional concrete materials to form an ultra high density concrete, DUCRETE, weighing more than 400 lb/ft³. DUCRETE can be used as shielding in spent nuclear fuel/high-level waste casks at a cost comparable to the lower of the disposal cost estimates. Consequently, the case can be made that DUCRETE shielded casks are an alternative to disposal. In this case, a beneficial long term solution is attained for much less than the combined cost of independently providing shielded casks and disposing of the depleted uranium. Furthermore, if disposal is avoided, the political problems associated with selection of a disposal location are also avoided.

Conceptual design studies have shown that a ventilated storage container for dry fuel storage similar to the Sierra Nuclear Corporation VSC-24 can be made from such ultra high density concrete. DUCRETE shielding results in a large reduction in both weight (35 tons) and diameter (40 inches smaller diameter) compared to conventional concrete casks.

Thus, a beneficial use has been found for the depleted uranium and the non-productive costs associated with disposal of the depleted uranium as a waste can be redirected to supporting nuclear utilities’ needs for spent fuel storage casks.

Other studies have also shown cost benefits for low level waste shielded disposal containers.

INTRODUCTION

As part of the Department of Energy’s Environmental Management (EM) Program, Office of Technology Development, in FY 1993 evaluation began on options for managing the DOE stockpile of over 400,000 metric tons of
depleted uranium (about 0.2% $^{235}$U) expressed as metal (Reference 1). Most of this material is stored as uranium hexafluoride (UF$_6$) at the gaseous diffusion enrichment plants. Additional uranium supplies exist in metallic and oxide forms elsewhere in the DOE complex.

The objective of the EM project was to evaluate management options by determining if there were any potential uses for the uranium and, if so, the associated costs. As a starting point, a disposal study was performed to establish a cost estimate for disposal of the uranium as radioactive waste (Reference 2). A summary of all EM-sponsored evaluations is provided in Reference 3.

**DISPOSAL OPTIONS**

The disposal study concluded that uranium hexafluoride would have to be converted to an oxide before disposal since it is chemically reactive with water. This conclusion is consistent with management practices in the French and British nuclear programs. Thus, the cost estimates were developed to include UF$_6$ conversion to U$_3$O$_8$, neutralization of the anhydrous hydrogen fluoride to calcium fluoride, packaging, transportation and disposal of U$_3$O$_8$ and CaF$_2$.

The DOE Nevada Test Site (NTS) and Hanford were considered in the disposal study since these sites are in an arid climate and probably most suitable for such large-scale disposal actions. (There was no effort to assess the technical viability of disposal at eastern DOE sites.) Disposal at private sites was considered, but deemed too costly. The study also considered the waste acceptance criteria at NTS and Hanford and modified the uranium oxide accordingly. At NTS, bulk oxide in metal containers was acceptable but at Hanford the oxide material required stabilization. Stabilization of uranium oxide was assumed for the study. It was estimated that stabilization in polyethylene would double the volume of waste disposed (This is a very optimistic loading and the volume of the mixture could easily go higher). The disposal unit cost was also different at each site and, over the duration of the study, changed at the NTS. The range of cost estimates for the baseline case with disposal at NTS and Hanford is presented in Table 1 (from Reference 3). This information has been adjusted from that presented in Reference 2.

**Table 1. Comparison of Disposal Costs for Baseline and Advanced Technology Disposal Options (375,000 MTU Basis)**

<table>
<thead>
<tr>
<th>Technology</th>
<th>Baseline Conversion</th>
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<td>Oxide LLW Disposal (NTS &amp; Hanford)</td>
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<tr>
<td>Oxide RCRA Disposal (Hanford)</td>
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<tr>
<td>Metal LLW Disposal (NTS)</td>
<td>$3.9</td>
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<td>New Plasma UF$_6$, Reduction</td>
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<td>Metal LLW Disposal (NTS)</td>
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<tr>
<td>Advanced UF$_6$, Conversion &amp; Elimination of CaF$_2$</td>
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<td>Oxide LLW Disposal (NTS &amp; Hanford)</td>
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<td>Oxide RCRA Disposal (Hanford)</td>
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<td></td>
</tr>
<tr>
<td>Advanced UF$_6$, Conversion to Oxide and U-Aggregate</td>
<td></td>
<td></td>
</tr>
<tr>
<td>LLW Disposal (NTS &amp; Hanford)</td>
<td>$1.6 to $1.9</td>
<td></td>
</tr>
<tr>
<td>RCRA Disposal (Hanford)</td>
<td>$2.4 to $3.4</td>
<td></td>
</tr>
</tbody>
</table>

**IMPACT OF ADVANCED TECHNOLOGY ON DISPOSAL COSTS**

The technology improvements leading to the greatest cost reductions are:

- processes for conversion of uranium hexafluoride to oxide with recovery of the hydrogen fluoride for recycle, and
densification of the oxide into depleted uranium aggregate.

Recovery and recycle of the hydrogen fluoride eliminates neutralization with calcium oxide and additional radioactive waste packaging, transportation and disposal costs. Although no contracts for such advanced conversion have actually been awarded, private sector companies have suggested that costs of less than half of that used in Reference 2 could be expected in a competitive proposal (Reference 4).

Disposal as a compressed, sintered aggregate rather than bulk oxide produces an estimated incremental savings of $300 to $600 million because of reduced disposal volume, which reduces the number of disposal containers and the disposal fees. Densification of the oxide powder into an aggregate reduces the cost of disposal by producing a waste form having a bulk density near 8 g/cm³. This compares to a bulk density of the powder of less than 3 g/cm³. Thus, disposal containers and disposal facility volume are reduced by about 60%.

Furthermore, the aggregate should be an acceptable waste form without further stabilization, resulting in even more savings at disposal sites requiring stabilization. Costs for large-scale production of the aggregate are unknown, but are anticipated to be small based on similar processes in the minerals industry.

DEPLETED URANIUM SHIELDING

After establishing the cost of disposal, a search for potential applications for the depleted uranium was made (Reference 3). The most viable use was determined to be for shielding of nuclear waste, especially spent fuel and high-level waste. Traditionally, uranium used in shipping cask shielding has always been metallic uranium. Costs to convert from uranium hexafluoride to uranium metal and then to fabricate to a final shape have been estimated to be near $20/kg (Reference 5). Only in special applications where cask weight or volume limitations demand the high gamma shielding efficiency of metallic uranium can such costs be justified. A metallic uranium spent fuel storage cask was estimated to be several times the cost of those of more conventional materials such as concrete (Reference 5).

Thus, either major cost reductions for UF₆ conversion to metal and metal fabrication or some other method would be required to use uranium in shielded containers. This situation gave rise to the concept of DUCRETE, which is compared to traditional concrete in Table 2.

<table>
<thead>
<tr>
<th>Concrete</th>
<th>DUCRETE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Type II Cement</td>
<td>Type II Cement</td>
</tr>
<tr>
<td>Sand (SiO₂ based)</td>
<td>Sand (SiO₂ based)b</td>
</tr>
<tr>
<td>Gravel (SiO₂ based)</td>
<td>Gravel (UO₂ based)</td>
</tr>
<tr>
<td>Density—150 lb/ft³</td>
<td>Density—400 to 450</td>
</tr>
<tr>
<td>Compressive Strength—3500 to 4000 lb/in²</td>
<td>Compressive</td>
</tr>
<tr>
<td></td>
<td>Strength—similar⁶</td>
</tr>
</tbody>
</table>

The key to making DUCRETE is the ability to fabricate a low-cost uranium oxide aggregate. (Note that this aggregate is the same as described above under the disposal cost reduction section. Thus, development of the aggregate has merits for both the disposal option and for use in DUCRETE.) Other chemical compounds of uranium such as uranium silicides were considered by the authors but rejected due to higher expected cost. Uranium oxide dissolved in

bConsideration has been given to using other sands such as colemanite and bauxite because of the presence of hydrogen in these minerals.

⁶Limited tests on DUCRETE showed lower strength (~20%), but this difference was attributed to the smooth surface of the "gravel" pellets (Ref. 9).
STRUCTURAL FEASIBILITY TESTS

Uranium dioxide is a relatively stable material and is considered essentially insoluble in water (Reference 6). Natural uraninite (UO$_2$) mineral deposits in contact with groundwater yield uranium concentrations from 50 to 200 ppb in the water (Reference 7). Thus, it was expected that uranium oxide aggregate would be chemically stable in the more basic environment of a Portland cement mixture (Reference 8). However, to verify this premise, depleted uranium oxide fuel pellets were obtained and a series of experiments were performed (Reference 9). These depleted uranium oxide fuel pellets were not the same aggregate that had been conceived for making DUCRETE, but were the only uranium oxide pellet material readily available within the DOE system. (Manufacturing a small quantity of depleted uranium aggregate for these proof-of-concept tests involved too much cost and lead time to be acceptable during the early phase of the project.)

To establish the initial feasibility of DUCRETE, Lessing (Reference 9) devised a series of experiments to test the physical characteristics of DUCRETE. Cement mixtures with varying concentrations of conventional crushed river aggregate (primarily quartz) and ceramic uranium oxide pellets were prepared. Samples from each mixture were fabricated for mechanical testing. The samples were prepared with careful attention to the exact composition, including water additions. A shear mixer was used to blend the ingredients.

Samples were cured per ASTM C 192, The Method of Making and Curing Concrete Test Specimens in the Laboratory. Compression strength for all samples were determined using ASTM C 39-72, Compressive Strength of Cylindrical Concrete Specimens. Brazilian tests were also performed to estimate the tensile strength of the specimens.

Samples of DUCRETE were formed with density as high as 400 lb/ft$^3$; conventional concrete density with the same volume percent gravel was
132 lb/ft$^3$. The samples were allowed to cure for times of 7, 28, and 90 days before testing.

These tests established that the mechanical properties of DUCRETE were similar to those of conventional concrete at the same volumetric loading of gravel. All samples of DUCRETE were above 3000 psi in the compression tests. In these tests, the DUCRETE samples displayed a slightly lower (~20%) strength than the gravel concrete samples. This result was attributed to the smooth surface of the fuel pellets used as aggregate. Having been made for other purposes, the pellets were centerless ground to a very smooth surface as required for nuclear fuel. This smooth surface causes very poor bonding of the cement fraction compared to the surface of crushed gravel. It is expected that DUCRETE fabricated with uranium aggregate that has rougher surfaces will behave similarly to crushed rock and will not display the lower strength observed in these tests. However, even these strengths are sufficient to be used in spent nuclear fuel casks (Reference 10).

**DUCRETE CASK DESIGN**

Since current spent nuclear fuel dry storage applications almost exclusively use concrete, it seems logical to substitute DUCRETE. Using low-cost aggregate fabrication methods from the minerals industry, it seemed feasible to produce the depleted uranium aggregate. Initial estimates were that DUCRETE with a density near 450 lb/ft$^3$ could be produced (a density of 400 lb/ft$^3$ was later confirmed by tests).

In parallel to the concrete tests described previously, several analytical studies were initiated to test the concept from a shielding performance perspective. INEL staff performed a variety of shielding calculations to determine if the postulated benefits of the high density concrete were suitable for source terms emanating from spent nuclear fuel. Additional studies were conducted by commercial firms in the shielded container and cask business.
In a later study, after some DUCRETE had been made, Hopf (Reference 12) determined that a ventilated storage container holding 24 PWR fuel assemblies could be designed with DUCRETE as the shielding material. This cask design was based on the Sierra Nuclear Company cask known as the VSC-24—a dry storage cask that uses natural circulation cooling to maintain fuel temperatures at suitable levels. An illustration of the VSC-24 is provided in Figure 3. Inside the VSC-24, a metal basket contains the spent fuel. This basket is dimensionally similar to the DOE Multi-Purpose Container.

The conceptual design of the DUCRETE VSC-24 is similar to the standard VSC-24 but is considerably smaller. Although the fuel load—24 PWR fuel assemblies—is the same, the external diameter is reduced by about 40 inches—from about 130 to 90 in.. The diameter varies somewhat depending on the assumed density of the aggregate and the volume fraction in the cement mixture. Hopf established a wall thickness range of about 8 to 12 in. for an aggregate density of 8 g/cm³ and various loading fractions. The total weight is also reduced, to about 100 tons from 135 tons. The design uses external and internal 0.5 in. thick steel shells with DUCRETE sandwiched between the shells. Nearly all other cask features are expected to be similar to the VSC-24.

This smaller diameter storage cask then stimulated the question of transportability since the diameter was well below the 128 inch limit for rail shipments. Could the DUCRETE VSC-24 with the loaded spent fuel storage basket be placed inside of a steel overpack and transported to a disposal facility? Hopf conducted a further study (Reference 13) that assessed this feasibility. The study used estimated thermal properties of DUCRETE, since no thermal performance data were yet available. This preliminary study indicated that transport inside of an overpack was possible, but would be limited by the heat generation rate of the fuel (age of the fuel since discharge from the reactor), the thermal performance of the DUCRETE cask, and heat rejection through the steel overpack. The concept also required a more sophisticated design for the DUCRETE storage container that used fins to conduct heat through the DUCRETE wall during transport. This design was estimated to be considerably more expensive than the basic DUCRETE storage cask design.

A further system benefits analysis by Powell (Reference 14) compared the economic performance of the VSC-24, the DUCRETE VSC-24, and the Transportable DUCRETE VSC-24. This study concluded that the transportable cask had no significant cost or performance advantages over the DUCRETE storage-only cask even if the thermal issues could be overcome. The DUCRETE cask compared well with the VSC-24 traditional concrete storage cask, showing a slightly lower life-cycle cost. Because of the lighter weight, the DUCRETE VSC-24 could be loaded directly in the fuel pools. This direct loading would reduce fuel handling costs and worker exposure.
ECONOMICS OF DUCRETE

The above referenced studies have established the technical feasibility of a spent fuel storage cask designed similar to the Sierra Nuclear Ventilated Storage Cask System. The economic viability is dependent primarily upon the cost of making uranium aggregate. In the cost studies, it has been assumed that the cost of conversion of uranium hexafluoride to uranium oxide could be ignored as that step would be necessary for any option available to DOE other than "store indefinitely".

Given depleted uranium oxide powder, the aggregate can be fabricated using bulk material processing operations such as briquetting or extrusion followed by sintering. The fabrication process must involve low-cost methods at every step or the uranium aggregate will be impractical for this application or for disposal. Processes are under development that show promise for producing the low-cost aggregate. Results from projects planned for FY 1996 should establish the fabrication cost range for the aggregate.

BREAK-EVEN COST The lowest cost of the disposal options considered (as presented in Table 2), $1.6 billion, equates to about $170,000 per 40 tons of uranium, which is the approximate mass of uranium needed for a DUCRETE VSC-24 cask. This disposal option assumes a low-cost process for conversion of the oxide into uranium oxide aggregate. Thus, if a cask can be made for this price, it is clearly a reasonable economic alternative to disposal of UF₆. As the cost of disposal increases, the allowable costs for fabricating DUCRETE casks increase accordingly.

Based on the estimate (Reference 3) for the future cost of large-scale conversion of UF₆ to oxide of $4.20 per kg U ($1.90 per lb-U), the 40 tons of uranium oxide will cost about $150,000 to convert to oxide. Thus, for the low-cost disposal option, only about $20,000 of avoided cost remains for construction of the cask at a break-even cost. This avoided cost represents the cost of containers, transportation, and disposal fees. However, as disposal costs increase (consistent with recent history), the break-even point increases. At the disposal cost estimate of $1.9 billion (the Hanford disposal option), there is about $50,000 avoided cost for fabrication of the cask.

The above discussion optimistically assumes that a disposal site is available with no protracted delay or other large costs beyond the disposal costs of $30 per cubic foot used at the NTS and $59 per cubic foot used at Hanford. History has shown this optimism to be unwarranted when disposal of large quantities of radioactive material are involved. Thus, the actual avoided cost clearly will be higher than estimated in these studies and the cost of the break-even option will exceed the stated values of $20,000 to $50,000.

TARGET FABRICATION COST Detailed design of a DUCRETE storage cask has not been performed and, thus, there is not a good cost estimate for the fabrication. However, it is expected that the cost will be near $100,000 based on similarities to existing concrete systems. If this estimate is correct, the casks would cost only about $50,000 to $80,000 more than the break-even cost.

SYNERGISTIC SOLUTION In the absence of a geologic disposal system to accept spent fuel, utilities have been faced with the prospect of providing their own spent fuel storage. Although DOE was to have accepted fuel by the year 1998, that outcome is doubtful. Consequently, utilities are acquiring dry fuel storage capability from firms specializing in such systems. These storage casks or vaults cost near $200,000 for the storage of 24 PWR assemblies or 65 BWR assemblies.

So, if the utilities are forced to store spent fuel in dry storage casks, nearly 9500 casks will be required—assuming the plants run to the end of their design life. Thus, at about $200,000 for each storage cask, the utilities face a $1.9 billion future cost for dry storage systems. (As of today, only a very small fraction of the 9500 casks have been fabricated.)

Consequently, it appears that if the UF₆ were
converted to DUCRETE casks, a significant cost savings could be achieved over the cost of solving the UF₆ disposal problem and the spent fuel dry storage problem independently. As the taxpayers and nuclear utility rate payers are usually the same people, the potential for substantial savings is a real benefit. To exercise this option, DOE would need to contract for a service that converted UF₆ and produced casks. The casks could then be sold or leased to utilities at a reduced cost compared to their current options using conventional materials.

LOW-LEVEL WASTE DISPOSAL CONTAINERS

Another application for DUCRETE shielding is low-level waste (LLW) that cannot be contact handled due to high levels of radioactivity. (In the United States, low-level waste does not necessarily have low activity.) At the DOE site at Fernald, Ohio, there are plans to vitrify wastes containing radium that, after vitrification, is estimated to have a surface dose near 1 rad.

Present disposal plans are considering a recycled concrete disposal box to provide the necessary shielding and strength for transportation. The box is projected to have a 6-in. wall thickness. For the same shielding effectiveness, the wall thickness using DUCRETE would be only about 2 in. This thin DUCRETE wall would not be suitable by itself for such a container due to structural limitations. However, DUCRETE could be used as a liner inside a steel box that satisfied the DOE and DOT structural requirements for a Type A container.

The advantage of this concept is that, for a fixed external volume, the DUCRETE-lined steel box would have a larger payload volume. Since most disposal costs for such waste are based on external volume, considerable cost savings are projected for a box with a larger usable internal volume. A cost model has been developed for determining the life-cycle costs of various shipping and container options (Reference 15). This model has been adapted to assess the DUCRETE box option. A comparison of a concrete box with 6-in. wall thickness and a steel box with a 2-in. DUCRETE lining is shown in Table 3. It is clear that a DUCRETE wall container has a cost advantage for low-level waste disposal applications when all elements of the life-cycle cost are considered.

RECYCLED METAL USE

For each spent fuel storage cask, approximately 20 tons of steel are used. If DOE establishes a recycle capability for the contaminated steel throughout the DOE complex, a considerable portion of such steel could be used in spent fuel casks provided the metallurgical quality of the steel meets the needed standards. Similar applications exist for LLW and HLW containers. Small quantities of fixed surface or internal contamination of the steel will make little difference to the external dose to radiation workers since the total external dose is dominated by radiation from the waste inside. Thus, additional savings may accrue if recycled steel is available for use in casks or LLW containers.

ULTIMATE DISPOSITION OF DUCRETE CASKS

The DUCRETE cask, constructed from depleted uranium, is slightly radioactive and, as such, must be disposed as radioactive waste at the conclusion of its life. Since the DUCRETE cask has a small
diameter, it can be shipped by rail and could be

| Table 3. Comparison of Concrete and DUCRETE Shielded Boxes for Disposal of Fernald Vitrified Waste |
|---------------------------------------------------|---------------------------------------------------|
| Description                                        | Portland Cement Concrete Construction | Steel Box with DUCRETE Shielding |
| Purchase Cost                                      | $3500                                  | $3500                             |
| Outside Dimensions                                 | 72"x54"x58"                             | 72"x54"x58"                       |
| Wall Thickness                                     | 6"                                     | Steel-0.15" Ducrete-2.2"          |
| Internal Volume                                    | 61.3 ft$^3$                            | 94.6 ft$^3$                       |
| Empty Weight                                       | 10371 lb                               | 10899 lb                          |
| Loaded Weight                                      | 19353 lb                               | 24772 lb                          |
| Number of Boxes Required                           | 3177                                   | 2057                              |
| Life-Cycle Cost ($/ft$^3$ of waste disposed)       | $172                                   | $110                              |

used by utilities as a Type A LLW container. This heavily shielded cask could be used to dispose of the activated core components from reactor decommissioning (piping, reactor vessel segments, mechanical components), resins, filters, sludges, etc. and any other Class B and greater wastes. The polyethylene high-integrity containers currently used require a concrete box overpack for burial in any low-level waste site to prevent subsidence. DUCRETE casks would satisfy this requirement (Reference 11). Conventional concrete casks cannot fill this function as their diameter is above that allowable for unrestricted rail transport.

Alternatively, the DUCRETE cask can be shipped to the geologic repository and disposed within the ample space available in the drifts not occupied by fuel. If there is no future repository, the uranium oxide aggregate, contained in a concrete matrix within a steel-walled container, is a very suitable disposal waste form and waste package.

**URANIUM AGGREGATE LEACH BEHAVIOR**

Recent tests on samples of the depleted uranium aggregate have shown very low uranium leach rates. The results are comparable to those obtained from iron-enriched basalt and are superior to those of borosilicate glass. Thus, the leaching of uranium after disposal in the DUCRETE container matrix would be minimal and much lower than for U$_3$O$_8$ disposed as bulk powder.

**CONCLUSIONS**

This paper has summarized the DUCRETE materials development and applications studies conducted for the Department of Energy, Office of Technology Development.

As a disposal waste form, depleted uranium aggregate was shown to reduce the cost of disposal of uranium oxide by $300 to $600 million, depending on the disposal location. If the depleted uranium aggregate is used in DUCRETE for spent fuel/high-level waste casks or for shielding in low-level waste containers, further large cost avoidance is achieved.

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$^d$ The cost for the concrete box has been used by FERMCO staff for life-cycle planning. It is assumed that the DUCRETE box can be fabricated for a similar cost.

$^e$ Per 10CFR61, B and C wastes must remain stable for 300 years. Class C and greater must be protected with intrusion barriers for 500 years.
These studies have shown that depleted uranium can be used for nuclear shielding applications and that significant performance and cost benefits are obtained compared to the disposal option.

Further work on the production-scale fabrication of depleted uranium aggregate is needed to improve cost estimates. Final design and demonstration of a DUCRETE spent fuel cask is needed to verify the technical feasibility discussed in this paper.

ACKNOWLEDGMENTS

This work was supported by the U.S. Department of Energy, Assistant Secretary for Environmental Management, under DOE Idaho Operations Office Contract No. DE-AC07-94ID13223.

REFERENCES

"White Paper"

SRS High Level Waste Canister Storage System Using DUCRETE Overpacks

Revised July 11, 1997

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Abstract

This paper identifies a concept for a low cost HLW storage method for the Savannah River Site DWPF canisters. The concept suggests a system solution where the surplus UO$_3$ and the contaminated stainless steel heat exchanger materials at SRS are recycled into DUCRETE Overpacks for interim storage of the DWPF canisters. A cost analysis is presented showing a lower cost storage concept compared to a second Glass Waste Storage Building. Also shown are the additional cost savings from the cost avoidance associated with disposal or indefinite storage of the SRS UO$_3$, and contaminated stainless steel materials.
High Level Waste Canister Storage System Using DUCRETE Overpacks

W. J. Quapp
Nuclear Metals, Inc.

1. Introduction:

The SRS has over 42,000,000 pounds of depleted uranium oxide (UO$_3$) stored in over 35,000 drums. Although the UO$_3$ is not a hazardous waste, the cost of storing that much nuclear material is never the less significant$^1$. What’s more, since there is presently no planned future use, the storage could be indefinite$^2$. One option available to the DOE at SRS is to use the depleted uranium oxide to produce DUCRETE™ Storage Overpacks for the vitrified HLW canisters$^3$ generated by the Defense Waste Processing Facility (DWPF).

This paper identifies a lower cost option for the storage of the vitrified DWPF HLW canisters at the Savannah River Site (SRS) using DUCRETE Concrete Shielding$^4$ and compares it to the current baseline cost of constructing an additional Glass Waste Storage Building. This concept also can incorporate a method for economically recycling the SRS contaminated stainless steel heat exchanger materials into metal components for the HLW overpacks and DWPF canisters.

Cost and benefit perspectives are provided on the total system costs from the UO$_3$ conversion to aggregate, recycling of the metal, fabrication of DUCRETE Overpacks, interim storage of the HLW on a secure concrete pad, and eventual disposition of the HLW and overpacks.

2. Background:

In 1993, the DOE Office of Science and Technology initiated a series of studies to evaluate the management options and liabilities for use or disposal of depleted uranium. A summary of those studies is provided in Depleted Uranium: A DOE Management Challenge, (Reference 1). Most of the focus was associated with evaluating the disposition of the uranium hexafluoride in the DOE inventory (representing more than 95% of the total). The depleted uranium (DU) inventory in 1993 consisted of about 550,000 metric tons of uranium hexafluoride stored in over 46,000 carbon steel cylinders and another 25,000 tons in metallic and other chemical forms. Conversion of the DU to a metal or oxide form was considered for the disposal options along with the possible use of DU metal or DUCRETE Concrete for shielding of HLW canisters and spent fuel storage containers. Cost estimates ranging from nearly $3 billion to $11 billion were associated with the various disposal options (Reference 2). The magnitude of this material management...

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$^1$ No storage cost at SRS has been available but at the Fernald DOE site, an estimate of annual storage costs for 8 million pounds of DU produced a cost of $0.0625 per pound per year. This cost applied to the SRS inventory would amount to nearly $3 million per year.

$^2$ The current SRS management plan for this material is Indefinite Storage — Savannah River Site Report, Depleted Uranium Disposition Study, NMP-92-06, Rev.1

$^3$ DUCRETE is a trademark of Lockheed Martin Idaho Technologies Company and is licensed to Nuclear Metals, Inc.

$^4$ This system would be deployed after the present Glass Waste Storage Building system is filled.
challenge is generally obscured by the many other more publicly sensitive issues at the major DOE sites.

A series of reports have been issued which describe the results of the studies conducted for DOE (References 3 to 10). One of the outcomes of these studies was the development of a material named DUCRETE Concrete (References 4 and 5). This is basically a combination of traditional Portland cement, sand, water, and an unusually high density gravel made from depleted uranium oxide. A low cost ceramic manufacturing process is employed to produce the aggregate. The mixture behaves like concrete but has the density (400 to 450 lb/ft³) nearly as high as steel.

DUCRETE Concrete can be used for shielding in HLW and spent fuel dry storage applications. Conceptual design studies for dry spent fuel storage overpacks have shown that both size and weight reductions can be expected (References 6 to 9). In applications which require neutron attenuation in addition to gamma shielding, the chemically bound water in the concrete provides excellent neutron shielding. The aggregate contains about 1% boron for manufacturing reasons but, if additional nuclear poisons (gadolina, hafnia, etc.) are desired, they can be incorporated into the aggregate for additional neutron attenuation or criticality control. Aggregate with a specific gravity of 8.75 gm/cm³ has been produced in laboratory tests. A pilot scale aggregate production facility (1 ton/day) has been installed at the NMI Carolina Metals Facility in Barnwell, SC. This facility is presently undergoing startup operations. A simplified process flowsheet for aggregate production is shown in Figure 1.

![Figure 1. Depleted Uranium Aggregate Simplified Process Flowsheet](image)

3. Alternative Shielding and Storage Option for SRS HLW

Currently, canisters holding the solidified high activity glass waste logs from the SRS Defense Waste Processing Facility are transported to a building designed and constructed for safely storing such canisters until they are shipped to a final geologic repository. This operating concept is illustrated in Figure 2. The present GWSB has sufficient capacity for storing
approximately one half of the projected number of canisters to be generated over the life of DWPF. Consequently, another facility is planned to provide the additional storage needs.

In Reference 10, Yoshimura reports on the results of a study performed for DOE which looked at an alternative storage concept where DWPF canisters are stored in overpacks made from a combination of steel and depleted uranium metal. After loading, the overpacks are stored on a concrete pad (similar to present practices for spent fuel at commercial utilities). This concept is illustrated in Figure 3. In this study, these depleted uranium metal overpacks were used to both provide interim storage and to transport the DWPF canisters to the repository. The lifecycle costs of the depleted uranium metal cask system were compared by Yoshimura against the present HLW baseline storage costs for SRS, West Valley and Hanford (Reference 10).

This paper presents the results of another HLW storage alternative using a depleted uranium overpack which is made from DUCRETE Concrete. The operating concept is similar to that illustrated in Figure 3 but does not use the DUCRETE overpack for transport of the DWPF.
canisters to the repository. The DUCRETE overpack system stores the HLW canisters in MPC canisters. The canisters are loaded into the DUCRETE overpacks which are stored on a concrete pad, illustrated in Figure 4, as in Yoshimura's study. This concept could also replace the SRS second Glass Waste Storage Building (GWSB) and at a lower cost by using materials now at the SRS.

Shipment of the HLW canisters to the geologic repository is performed using conventional licensed spent fuel transportation casks. The DUCRETE overpack is shipped by rail empty to the geologic repository for ultimate disposition. Depending upon the requirements for a disposal package for the HLW, the canisters could be reinserted into the overpack at the repository site and disposed to provide a shielded disposal option. Alternatively, the canisters are disposed elsewhere in the geologic repository separate from the HLW disposal package.

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5 The MPC is a multipurpose canister developed by the DOE Office of Civilian Radioactive Waste Management so that a standard mechanical interface was used for material to be disposed in the planned geologic repository.
This study uses data compiled by Yoshimura for the capital, operating and decommissioning cost of the second SRS GWSB facility as well as the costs for the overpack storage pad and the associated handling infrastructure. These costs are used to develop a baseline life cycle cost for an alternative DUCRETE overpack scenario for the DWPF canisters.

Baseline costs for the SRS Glass Waste Storage Building are listed in Table 1. The important element of this table is the total unit cost per canister for this storage mode -- $66,000. The costs for the Pad Based Storage System are presented in Table 2 (Reference 10). For this option, the unit cost per canister is $6600, however, this cost does not include the storage overpack. In Table 3, the cost comparison between the GWSB and DUCRETE overpack options is presented. Since detailed design of the overpack has not been completed, a range of overpack costs is represented to reflect the uncertainty in this cost element. This analysis shows that a cost saving

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6 Costs are slightly higher than Yoshimura’s ($4900) due to a different basis for this study. Also, a recently acquired developed for the DOE OCRWM program has capital cost estimates for the pad infrastructure which are approximately $3200 per overpack which would equate to $800 per DWPF canister. This “White Paper” will be updated to reflect this apparently lower cost after it is validated to include the same cost basis.
somewhere between $35 million may be attainable using the DUCRETE overpack option compared to another GWSB.

Table 1. SRS Glass Waste Storage Building' (Costs in Millions)

<table>
<thead>
<tr>
<th>Component</th>
<th>Capital Cost</th>
<th>Decommissioning @ 20% of Capital</th>
<th>Annual Operating</th>
<th>Total Operating For 20 years</th>
<th>Total Life Cycle Cost</th>
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</thead>
<tbody>
<tr>
<td>GWSB</td>
<td>$111</td>
<td>$22.2</td>
<td>$0.75</td>
<td>$15</td>
<td>$148.2</td>
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<tr>
<td>Shielded</td>
<td>$20.6</td>
<td>$4.1</td>
<td>Included in</td>
<td>Included</td>
<td>$24.7</td>
</tr>
<tr>
<td>Canister Transporter</td>
<td></td>
<td></td>
<td>Above</td>
<td>in Above</td>
<td></td>
</tr>
</tbody>
</table>

Total Life Cycle Cost $172.9

Total Lifecycle Cost $0.066 Per Canister


Table 2. SRS Glass Waste Canister Storage Pad' (Costs in Millions)

<table>
<thead>
<tr>
<th>Component</th>
<th>Capital Cost</th>
<th>Decommissioning @ 10% of Capital</th>
<th>Annual Operating</th>
<th>Total Operating For 35 years</th>
<th>Total Life Cycle Cost</th>
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<tbody>
<tr>
<td>Storage Pad</td>
<td>$6.8</td>
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<td>$0.12</td>
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<td>$11.6</td>
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<tr>
<td>2 Mile Road</td>
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<td>$0.024</td>
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<td>Heavy Hauler2</td>
<td>$0.25</td>
<td>$0.025</td>
<td>$0.09</td>
<td>$3.15</td>
<td>$3.42</td>
</tr>
</tbody>
</table>

Total Life Cycle Cost $17.4

Total Lifecycle Cost $0.0066 Per Canister

1. Capacity of about 2620 canisters. Data from Reference 10, Appendix D, Table D-30. Capacity related costs divided by two to reflect reduced capacity requirements compared to original study.
2. One Heavy Hauler required. Operation costed for a 35-year period.
3. Differs from Reference 3 because of higher fixed cost for this one storage facility capacity.

Table 3. Comparison of SRS Additional GWSB Versus DUCRETE Overpack Storage

<table>
<thead>
<tr>
<th>Cost Element</th>
<th>Storage Building</th>
<th>DUCRETE Overpack Storage</th>
</tr>
</thead>
<tbody>
<tr>
<td>Unit Cost Per Canister</td>
<td>$66,000</td>
<td>$6,600</td>
</tr>
<tr>
<td>Number of Canisters</td>
<td>2620</td>
<td>2620</td>
</tr>
<tr>
<td>Canisters per Overpack</td>
<td>NA</td>
<td>4</td>
</tr>
<tr>
<td>Number of Overpacks</td>
<td>NA</td>
<td>655</td>
</tr>
<tr>
<td>Cost per Overpack</td>
<td>NA</td>
<td>$200,0001</td>
</tr>
<tr>
<td>Cost of Overpacks</td>
<td>NA</td>
<td>$131,000,000</td>
</tr>
<tr>
<td>Rail Transport Cost to Geologic Repository</td>
<td></td>
<td>$4,000</td>
</tr>
<tr>
<td>Total Cost Per Canister</td>
<td>$66,000</td>
<td>$52,650</td>
</tr>
<tr>
<td>Total Life Cycle Cost For Storage System</td>
<td>$172,920,000</td>
<td>$137,943,000</td>
</tr>
<tr>
<td>Potential Savings</td>
<td>NA</td>
<td>$35 million</td>
</tr>
</tbody>
</table>

1. Estimated overpack cost including cost of depleted uranium aggregate production from government furnished oxide.
4. Avoided Costs

There are two kinds of avoided costs that can be taken as "credit" in computing the total system cost of this proposed option. These include:

- Costs for managing the SRS UO3
- Costs for managing the SRS contaminated stainless steel heat exchangers

Approximately 45 tons of uranium aggregate (at 90 w% loading of UO2) and about 9 tons of stainless steel will be used per overpack. Based on a simplified conceptual design as shown in Appendix A, the 655 DUCRETE overpacks would consume 28,000 tons of UO2 and 5900 tons of steel. The uranium oxide content exceeds the SRS inventory of 21,000 tons of UO3, however, the additional uranium material needed can be obtained elsewhere in the DOE Complex. The steel content nearly equals the available steel tonnage in the contaminated stainless steel heat exchangers at SRS. Both of these additional avoided costs are discussed in more detail below.

4.1 Cost Avoidance from Using the UO3 Material

The canister cost used in the above analysis presumed the depleted uranium aggregate was available as government furnished material. However, the cost of this conversion to aggregate is an additional cost. The overpack conceptual design contains approximately 30 tons of aggregate in 10.7 yd3 of DUCRETE Concrete with a calculated density7 of 395 lb/ft3. Higher density DUCRETE Concrete may be achievable which will allow some additional design optimization.

As "Owner" of the UO3, the SRS has adopted a management position of Indefinite Storage8. The cost of this "indefinite option was identified by SRS to be $1.13 per pound. The cost of converting the depleted uranium to an aggregate was estimated by NMI to be between $0.57 to $0.72 per pound depending upon time for conversion (3.5 to 7 years). This cost also assumes that the facility is used only to process the SRS inventory (approximately 21,000 tons or 42 million pounds)9. Increasing processing quantities will lower unit costs. Depending upon DOE policy, this is either an added cost or a reduced liability. If an added cost, the aggregate production will cost about $27 million using the mean value of the unit production cost. Alternatively, compared to the finite cost of $1.13 for indefinite storage, the $0.485 per pound difference (mean value) would produce an additional savings of about $20.4 million.

7 Calculation of the density of the concrete is uncertain depending upon the assumptions made. Values between 395 to 480 lb/ft3 can be estimated depending on modeling assumptions.
8 Savannah River Site Report, Depleted Uranium Disposition Study, NMP-92-06, Rev.1
9 Cost estimate based on facility conceptual design and estimated staffing requirements for aggregate production. Facility located at CMI in Barnwell, SC.
4.2 Cost Avoidance From Heat Exchanger Metal Recycle

NMI has investigated the application of the OSPREY metal spray process for production of steel cylinders for these overpacks. This system uses a molten metal feed and sprays it onto a rotating mandrel to build up a cylinder of the appropriate length and thickness. Where tight tolerances dictate, the cooled part is machined to the finished dimension. The process has been deployed at many places around the world including in the US for the Navy's Submarine program for production of heavy wall pipe. Based upon discussions held between NMI and the OSPREY organization in England, the production cost of large diameter cylinders is expected to be similar to the costs of conventional alternative metal processing. However, unique to the OSPREY process is the ability to use scrap metal as feedstock. Thus, this provides an opportunity to use the SRS heat exchangers in a real application on a cost-effective basis as an alternative to burial or use at unrealistic costs.

A rough estimate of the metal content for the overpack, the MPC, and the four HLW canisters has been made. The conceptual design for this system is shown in Figure 5. DWPF canister weight of 1100 pounds was used. The inner and outer overpack shells are made from stainless steel and the HLW canisters are loaded into a larger canister similar to the OCRWM MPC canister. The MPC is loaded into the overpack. The rough order of magnitude weight of the steel elements of this system yielded 5.5 tons of steel for the shells surrounding the overpack, 1.5 tons for the MPC shell, and an additional 2.2 tons for the 4 HLW canisters. Thus, the total metal mass per overpack including the 4 HLW canisters is approximately 9.2 ton. This metal recycle represents an avoided burial cost of about $41,000 per overpack using $2.22 per pound of metal for disposal. For 655 overpacks, the total cost savings reaches about $27 million. The total metal used in this conceptual design scenario is about 6000 tons compared to the SRS inventory of about 7000 tons.

---

10 Telecon between W. Boettinger, SRS, and W. Quapp, NMI, April 29, 1997
4.3 Net Overpack Cost after Avoided Cost Credits

The net overpack costs is computed by combining all estimated costs and the cost avoidance. The estimates are provided in Table 4. These are illustrative costs and more detailed design efforts are needed to improve the available cost savings.

Table 4. Net Overpack Cost After Cost Credits

<table>
<thead>
<tr>
<th></th>
<th>Cost/Credit Estimate</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial Overpack Cost</td>
<td>$200,000</td>
</tr>
<tr>
<td>Cost of Aggregate</td>
<td>$55,000</td>
</tr>
<tr>
<td>UO$_2$ Cost Credit</td>
<td>-$41,000</td>
</tr>
<tr>
<td>Metal Recycle Cost</td>
<td>-$41,000</td>
</tr>
<tr>
<td>Final Net Cost Per Overpack</td>
<td>$173,000</td>
</tr>
</tbody>
</table>

4.4 Total Cost Savings

This section summarizes the total cost savings from this combined strategy of using overpacks fabricated from SRS surplus materials for HLW storage. The data are shown in Table 5. Although there is uncertainty in these costs, the potential for large savings is clearly evident.

Table 5. Total Cost Savings From Deployment of Overpack Storage Using SRS UO$_2$, and Contaminated Steel Materials

<table>
<thead>
<tr>
<th></th>
<th>Cost</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial Cost of Overpack</td>
<td>$200K</td>
</tr>
<tr>
<td>Compared to GWSB</td>
<td>$35M</td>
</tr>
<tr>
<td>Savings From UO$_2$ Use</td>
<td>$20 M</td>
</tr>
<tr>
<td>From Metal Recycle Use</td>
<td>$27 M</td>
</tr>
<tr>
<td>Total Estimated Savings</td>
<td>$82M</td>
</tr>
</tbody>
</table>

5. Cost Risks

The current state of cask design is very premature and the range of costs was chosen to be representative of commercial spent fuel cask overpacks. A detailed design effort is needed to firm up these estimates. However, there are also many scenarios that can make the UO$_2$ management cost increase significantly. In particular, the USNRC, in a licensing action for the Louisiana Enrichment Facility, has suggested that depleted uranium should be converted to an oxide and disposed in a deep mine rather than shallow land disposal (Reference 11). U$_3$O$_8$ was chemical form suggested by the NRC, however, the SRS UO$_2$ material should be suitable and meet the intent of the NRC letter. In the SRS management options study previously cited, the cost of geologic disposal was estimated at $3.85 per pound ($162 million)$^{11}$.

Thus, if DUCRETE Overpacks are successfully used and disposed with the vitrified waste in the future spent fuel geologic repository, this complex and expensive UO$_2$ management issue may be solved at a relatively low life cycle cost.

6. Other Benefits

The other benefits to the SRS fall into two categories: 1) local jobs creation, and 2) better environmental solutions.

$^{11}$ Savannah River Site Report, *Depleted Uranium Disposition Study*, NMP-92-06, Rev.1
**Jobs Creation:** Since the UO₂ and the heat exchanger material are located on the SRS Site, it would make economic sense to perform the material conversion and overpack fabrication as close as possible to the SRS site to minimize unnecessary shipping costs. The Nuclear Metals facility – Carolina Metals, Inc. – is located adjacent to the SRS site in Barnwell, SC. This 270-acre site contains two buildings that have been used for depleted uranium manufacturing activities since 1983. This site is ideal for such an undertaking a depleted uranium aggregate production operation. The metal recycle operation could be performed in one of the SRS surplus facilities such as Building 320. Cask assembly might be done at either location depending upon final facility design. Once the precedent for using DUCRETE Overpacks for HLW are successfully demonstrated, future manufacturing of DUCRETE spent fuel storage casks could follow. Conceptually, over 9000 casks could be required for spent fuel over the next 30 years. This manufacturing activity could provide the long-term jobs creation needed in this area of the country.

**Better Environmental Solution:** By converting the UO₂ into an aggregate, combining it in concrete which is poured between steel shells, the quality of the end product represents an excellent environmental waste form at the end of the useful life of the storage overpack. Leach tests using ANS 16.1 procedures has shown excellent performance with “non-detect” results for uranium in the leachate. Consequently, the long-term behavior in a future repository should be excellent and not add a substantial burden upon the repository environmental risk.

7. Schedule and Budget Impact

A top-level conceptual schedule is shown in Figure 5. Most importantly, by beginning this work soon, there is sufficient lead-time for construction of an aggregate production facility, a metal recycle facility, cask design and licensing, and cask fabrication facility before the overpacks are needed for HLW. Costs to DOE during the first several years will be relatively low during facility construction and initial operations as private sector financing will be used to construct facilities, thus allowing for the orderly time phasing and redirection of funds for this project. Additionally, since the storage capacity is procured over the operating term of the DWPF, most of the costs can be extended over this time frame. This concept provides lower total costs than the GWSB option requiring up front costs for facility construction. This funding approach reduces the net present value of the overpack.
8. Conclusions

The combined needs for shielded storage, UO₃ disposition, and contaminated metal disposal are best addressed through an integrated system engineering solution. Such a solution has been identified in this paper and preliminary estimates of the potential cost savings have been prepared. Based on the analysis presented in this paper, the use of a canister overpack system should be seriously considered for storage of the SRS vitrified HLW after the capacity of the present Glass Waste Storage Building system is fully used. Major cost savings have been estimated which can easily exceed $80 million and may be higher. The overpack concept will not only have lower life cycle costs than a second GWSB, but it will use the excess depleted uranium and the contaminated heat exchanger material at the SRS and provide additional management flexibility. Lastly, from a local interest perspective, jobs are created in the SRS community in lieu of spending funds on trucking contracts and disposal fees elsewhere in the country.

9. References:


“White Paper”

Options For Using or Disposing of Depleted Uranium in the DOE Geologic Repository

Introduction:

This document provides a brief description of options considered by the EM 50 and NE task force members and provides a subjective assessment of their advantages and disadvantages.

This paper assumes that the reader understands DOE’s responsibility to manage and eventually dispose of nearly 600,000 metric tons of depleted uranium hexafluoride. All options assume the hexafluoride is converted into a stable oxide before re-use or disposal.

If the option appears to fail to be viable, it is so stated. This document is intended to represent a brainstorming effort and the suggestion and evaluation of an option does not imply that the authors believe it necessarily a good idea. This document is merely presented as an options list. This paper has been prepared to identify those options considered by the task force members for the benefit of workshop attendees and to document the rationale behind them.

This paper uses considerable subjective judgement. Information may be available to others that would change some of these judgements relative to viability. An option would not necessarily be omitted because of incompatibility with the current planning basis for the repository.

General Comments on All Options:

1) Virtually all options will consume all or most of the DOE depleted uranium inventory. Actual quantity required is a function of detailed design, implementation schedule, and other variables.

2) All options increase the heavy metal inventory in the repository but the final form (powdered oxide, aggregate and metal) differs depending on application. The final form can make a difference on dissolution of the uranium oxide in ground water and subsequent transport to a potential receptor. This may have an impact on the performance assessment at some future time.

3) Any DU handling at the repository may have some radiological impact on workers but for options such as casks where the DU is encapsulated inside of steel shells, this impact is minimal. Previous estimates of dose penetrating the DUCRETÉ cask steel wall were on the order of $\frac{1}{2}$ mR/h. Options that handle bulk uranium oxide or uranium aggregate materials are quite a different story, as there is a potential for inhalation doses. However, because of the stabilized form of the depleted uranium aggregate, radiological management should be much easier than managing bulk oxide.
1. DUCRETE Casks for At-Reactor Storage and Disposal of DUCRETE Casks (w/o Fuel) in Empty Buffer Drifts

OPTION DESCRIPTION: This option uses the DOE depleted uranium hexafluoride by first converting it to an oxide that is subsequently manufactured into a ceramic aggregate. The aggregate is subsequently used in DUCRETE concrete dry fuel storage casks. Replacing the conventional aggregate in concrete with a heavy aggregate made from depleted uranium oxide makes DUCRETE concrete. The DUCRETE concrete has an estimated density of over 400 lb/ft$^3$. This high-density concrete provides excellent gamma ray attenuation and the hydrogen in the bound water provides neutron attenuation. The DUCRETE Concrete can be used in place of ordinary concrete for storage casks. This concept is the same as conventional concrete casks produced by dry fuel storage cask vendors such as Sierra Nuclear, NAC, and Holtek. Because of the higher density, DUCRETE Casks are smaller diameter and weigh less than casks made from ordinary concrete. These attributes make transportability possible. Conventional concrete casks are manufactured on the utility site and used there with no provision for transport elsewhere because they exceed the width requirements for unrestricted rail transport. Comparative diameters of a conventional and DUCRETE Concrete cask are about 135 inches and 90 to 100 inches, respectively. DUCRETE cask are also estimated to weigh about 35 tons less.

The DUCRETE Casks can be used at the utility sites for interim storage, they can be shipped from the utility to the DOE Interim Storage facility when it becomes available, and finally shipped to the geological repository for lag storage when it is available to receive fuel. The DUCRETE Casks are not used to ship fuel but only to store it at the storage sites. If not needed for lag storage at the repository, it can be re-deployed at a utility sight or at the DOE Interim Storage Facility. When no longer needed for fuel storage, the casks may be disposed in the repository by using any remaining buffer drifts and excavating additional drift space as required. As there is essentially no heat load from the depleted uranium in the DUCRETE Cask, these casks can be packed as close as physically possible.

OPTION ADVANTAGES:

- Provides for a lower-cost dry fuel storage system since the same casks are used at the utilities and the Interim Storage Facility.
- May reduce the cost of fuel storage at the repository.
- Provides for final disposal in an environmentally excellent final waste form.
- Minimizes the cost of disposal and eliminates the need for a new geologic disposal facility.
- Utilizes existing repository drift space and reduces backfill closure work for these drifts.

OPTION DISADVANTAGES:

- May require the excavation of additional drift space at repository.

VIABILITY ASSESSMENT: Yes
2. DU Metal Casks for At-Reactor Storage and Disposal of DU Metal Casks (w/o Fuel) in Empty Drifts

OPTION DESCRIPTION: This option is essentially similar to option 1. Only the differences will be highlighted. DU Metal has been used for shielded cask for many years. This concept uses DU Metal sandwiched between stainless steel shells. The stainless steel provides the required strength as DU Metal, while strong, is not an ASME Code material. It requires no development effort, as does DUCRETE Concrete. Depleted uranium metal has been used in this type of application for licensed transportation casks for many years. However, the cost to produce DU Metal is many times higher than estimated for DUCRETE Concrete. Consequently, casks would be much more expensive to fabricate than currently estimated for DUCRETE Concrete. It is expected that the same types of cost differences between DUCRETE and DU Metal would exist as between conventional metal and concrete casks. An advantage of DU Metal casks over DUCRETE is the thermal conductivity of the material would preclude the need for natural circulation venting, as is the case for concrete systems. The DU Metal casks would be lighter and smaller than the DUCRETE Concrete casks, however, this feature may be of little value as both are rail transportable.

Disposal of DU Metal casks does not meet previously identified NRC requirements for disposal as an oxide. However, this disposal mode should be environmentally acceptable. The only known issue may be associated with the future potential corrosion of the depleted uranium with water and the resultant formation of hydrogen gas. This issue could probably be mitigated by repository design.

OPTION ADVANTAGES:

- May reduce the cost of fuel storage at the repository.
- Uses the DOE depleted uranium and provides for final disposal.
- Minimizes the cost of disposal and eliminates the need for a new geologic disposal facility.

OPTION DISADVANTAGES:

- Requires the excavation of additional drift space at repository.
- Higher cost than conventional concrete casks, conventional metal casks, and DUCRETE Concrete casks.
- Unknown impact on repository performance due to corrosion of depleted uranium metal after corrosion failure of the stainless steel shell.

VIABILITY ASSESSMENT: Yes
3.0 DUCRETE Casks for Dry Fuel Storage and Self-Shielded Disposal Packages.

OPTION DESCRIPTION: This option is identical to Option 1 except for the final disposal in empty drift space. This option brings each DUCRETE cask to the repository. The fuel canister is fitted with the corrosion resistant disposal package and reinserted into the DUCRETE Cask (The cask would be oversized when fabricated for eventual insertion of the disposal package at the repository.). The cask and its contents are then moved into position in a suitable location underground using a horizontal rail car system. Since the packages are fully shielded, no remote operations are required. Due to the overall package size, some drift modifications may be necessary or desirable depending upon the emplacement method (rail versus gantry). Alternative concepts for disposal of the shielded package are also possible such as the mining of a large room for disposal in tightly packed groups. If horizontal disposal in drifts were selected, modifications to the natural convection cooling of the casks would be required during cask design (current storage casks use vented cooling designs for cask vertical orientation).

OPTION ADVANTAGES:

- Same advantages as Option 1.
- Required ventilation reduces the drift and fuel temperature, humidity, and simplifies operations during the emplacement period. At lower temperature and humidity, the material corrosion rates are dramatically reduced during the ventilation period. Uncertain behavior of the mountain as a result of the heating of the rock is reduced during ventilation period.
- Eliminates requirement for repository remote operations, lowers costs, simplifies design, and enhances maintainability.
- Simplifies licensing, reduces operational risks during emplacement period.
- Simplifies fuel retrieval and/or accessibility if needed to accommodate disposal package failure, performance verification and testing, and future recovery for enhanced disposal using new technical developments.
- Defers closure for a long period (100 years).

OPTION DISADVANTAGES:

- Requires the drifts to be ventilated to maintain acceptable fuel temperatures.
- Increases cost following emplacement period.
- Defers closure for a long period (100 years).
- Simplifies future accessibility from a material safeguards perspective.

VIABILITY ASSESSMENT: Yes
Disposal Concepts for DUCRETE or DU Metal Storage Casks

Self-Shielded Disposal Overpacks
Using DUCRETE Concrete Overpacks In Drifts

Self-Shielded Disposal Overpacks
Using DUCRETE Concrete Overpacks In "Super-Drift"
4. DU Metal Casks for Dry Fuel Storage and Self-Shielded Disposal Packages.

OPTION DESCRIPTION: This option is identical to Option 2 except for the final disposal in empty drift space. This option brings each DU METAL cask to the repository. The fuel canister is fitted with the corrosion resistant disposal package and reinserted into the DU METAL Cask (The cask would be oversized when fabricated for eventual insertion of the disposal package at the repository.). The cask and its contents are then moved into position in a suitable location underground using a horizontal rail car system. Since the packages are fully shielded, no remote operations are required. Due to the overall package size, some drift modifications may be necessary or desirable. Alternative concepts for disposal of the shielded package are also possible such as the mining of a large “room” for disposal in tightly packed groups.

OPTION ADVANTAGES:

- Same advantages as Option 2.
- Required ventilation reduces the drift and fuel temperature, humidity, and simplifies operations during the emplacement and ventilation period. At lower temperature and humidity, the material corrosion rates are dramatically reduced during the ventilation period. Uncertain behavior of the mountain as a result of the heating of the rock is reduced during ventilation period.
- Eliminates requirement for repository remote operations, lowers costs, simplifies design, and enhances maintainability.
- Simplifies licensing, reduces operational risks during emplacement period.
- Simplifies fuel retrieval and/or accessibility if needed to accommodate disposal package failure, performance verification and testing, and future use or recovery for enhanced disposal using new technical developments.
- Defers closure for a long period (100 years).
- Heat removal from the DU METAL cask is simpler than for a DUCRETE Cask since it does not depend upon internal natural convection.

OPTION DISADVANTAGES:

- Requires the drifts to be ventilated to maintain acceptable fuel temperatures.
- Increases cost following emplacement period.
- Defers closure for long period (100 years).
- Simplifies future accessibility from a material safeguards perspective.
- Higher cost than conventional concrete casks, conventional metal casks, and DUCRETE Concrete casks.
- Unknown impact on repository performance due to corrosion of depleted uranium metal after corrosion failure of the stainless steel shell.

VIABILITY ASSESSMENT: Yes
5. DUCRETE Inverts in Repository Drifts

OPTION DESCRIPTION: This option simply uses the depleted uranium aggregate developed for DUCRETE Concrete and substitutes for the conventional aggregate in the manufacturing of drift inverts.

OPTION ADVANTAGES:

- Provides for final disposal in an environmentally excellent final waste form.
- Provides some advantage for the dilution of fissile material to reduce hypothetical risk of future criticality.

OPTION DISADVANTAGES:

- Worker radiological exposure in fabrication of inverts (Might be done offsite in a controlled environment to minimize contamination risks).

VIABILITY ASSESSMENT: Yes
6. DUAGG as Backfill Around Disposal Package

OPTION DESCRIPTION: This option simply uses the depleted uranium aggregate developed for DUCRETTE Concrete and deposits it around the disposal package for dilution of the fissile uranium in the distant future. This option could use all of the DOE depleted uranium.

OPTION ADVANTAGES:

- Provides for a depleted uranium disposal option.
- Converts the depleted uranium into a very environmentally stable form.
- Provides some advantage for the dilution of fissile material to reduce hypothetical risk of future criticality.

OPTION DISADVANTAGES:

- Emplacement method will require remote operations (required for any backfill operation).
- Potentially large radon release term

VIABILITY ASSESSMENT: Yes
7. DU Oxide as Backfill Around Disposal Package

OPTION DESCRIPTION: This option is similar to Option 6 except that uranium oxide is deposited directly around the disposal package for dilution of the fissile uranium in the distant future. This option could use all of the DOE depleted uranium.

OPTION ADVANTAGES:

- Provides the lowest cost depleted uranium disposal option.
- Converts the depleted uranium into a reasonably environmentally stable form.
- Provides some advantage for the dilution of fissile material argument to avoid recriticality.

OPTION DISADVANTAGES:

- Emplacement method will require remote operations (required for any backfill operation).
- Will represent a major personnel health hazard during any future drift ventilation period due to entrainment of the uranium oxide into the ventilation air.
- Small oxide particles are more soluble than aggregate and will increase the source term to receptors.
- Potentially large radon release term

VIABILITY ASSESSMENT: Marginal due to potential worker health risks and impact on PA
8. DU Oxide or DUSCOBS as Filler Inside of Fuel Canister

OPTION DESCRIPTION: This option uses depleted uranium in an oxide or small glass particle geometry. The material is poured into the canister at the repository. Conceptually, the purpose is to provide enhanced self shielding of the fuel canister and to provide material in intimate contact with the fuel to reduce risks of future criticality during postulated flooding events and in the post disposal package failure time when the fuel elements are predicted to be dissolved by groundwater.

OPTION ADVANTAGES:

- Uranium is contained in the disposal package.
- Probably lowest cost method of disposing of uranium in repository.
- Reduced personnel health risks compared to Option 7.
- Should provide improvement in crush resistance which is important for accident analysis.

OPTION DISADVANTAGES:

- Deleterious effect on early life, fuel element temperatures due to impeded heat transfer.
- Increases cost of fuel canister handling at repository.
- Increases contamination of fuel handling facilities and disposal canister.
- After disposal package failure, more mobile source term.
- Increased mechanical loading and stress on the fuel assemblies and fuel rod cladding creating adverse and unknown conditions.

VIABILITY ASSESSMENT: Questionable viability due to heat rejection through the oxide or glass material. Impact of mechanical stresses on the fuel rods is unknown. Feasibility of fully loading all voids is also unknown.
9. DUCRETE Shield Blocks Positioned Next to Unshielded Disposal Package

OPTION DESCRIPTION: This option uses DUCRETE Concrete to fabricate shield blocks for use in the repository. One postulated use would be to surround the emplaced disposal packages. If suitably surrounded, future operations and maintenance might be simplified. Alternatively, shield blocks (in a box like geometry) could accompany the disposal canister during emplacement and provide for partial shielding. Ventilation requirements would preclude complete encasement. DUCRETE Concrete shield blocks would be about 1/3 the thickness of conventional concrete shield blocks. Blocks here could include small walls and other useful shapes.

OPTION ADVANTAGES:

- Uranium is contained in the DUCRETE Concrete in environmentally excellent final form.
- Provides potential flexibility of operations.
- Block concept would use simple, standardized shapes.
- Dose reduction is space opposite disposal package might be reduced enough to simplify future backfill operations.

OPTION DISADVANTAGES:

- Remote emplacement techniques required for most concepts (except box).
- Radiation shine to ceiling of drift may provide only marginally effective personal protection.
- Cooling requirements may preclude effective shielding geometry.

VIABILITY ASSESSMENT: Yes, but uncertain effectiveness.

Self-Shielded Disposal Overpacks
Using DUCRETE Concrete Overpacks In Drifts
10. DUCRETE Half-Cylindrical Shield With End Ventilation

OPTION DESCRIPTION: This option is a variation of Option 9. It uses a DUCRETE half cylinder (Cylinder sliced axially through the centerline.) for shielding emplaced disposal package. Concept involves casting a depression in the invert blocks for accommodating some portion of the cylinder. Pre-cast DUCRETE Concrete blocks might provide a portion of the disposal package sidewall height. Finally, a half-cylinder would be emplaced covering the remainder of the disposal package. Ventilation air would be required to remove heat through end vents.

OPTION ADVANTAGES:

- Uranium is contained DUCRETE Concrete.
- Provides a shielded operating environment after placement of the blocks and half-cylinders.
- No reflective radiation dose from ceiling of drift as in Option 9.
- Greatly simplifies addition of backfill in drifts.

OPTION DISADVANTAGES:

- Complex, remote handling equipment need for emplacement of blocks and half-cylinders.
- Cooling airflow required for heat removal.

VIABILITY ASSESSMENT: Questionable feasibility due to requirements on emplacement and technical effectiveness.

Alternative DUCRETE Shield Block Configuration

![Diagram of Alternative DUCRETE Shield Block Configuration]
11. DUCRETE Cask Integrated with Disposal Package

OPTION DESCRIPTION: This option would integrate the disposal package into the design of the dry fuel storage cask. In a normal concrete storage cask, up to 1.75 inches (45 mm) steel lining is used to provide structural strength to the cask and to diffuse the heat load away from the concrete to prevent localized overheating. If the disposal package were to replace this cask component, total system weight and size reductions might be possible. Total system shielding requirements can be met at lower total weight than for Option 1. Complicating this concept is the need for ventilation.

OPTION ADVANTAGES:

- System concept minimizes overpack weight.
- Provides for a lower-cost dry fuel storage system since the same casks are used at the utilities and the Interim Storage Facility.
- May reduce the cost of fuel storage at the repository.
- Provides for final disposal in an environmentally excellent final waste form.
- Minimizes the cost of disposal and eliminates the need for a new geologic disposal facility.

OPTION DISADVANTAGES:

- Ventilation requirements cannot be met while maintaining a sealed disposal package unless a removable vent plug concept is acceptable for the corrosion resistant materials. Plug would be removed during all fuel storage phases up to closure of the repository.
- Requires expensive disposal package early in life of program, thus, increasing net present value of cost.

VIABILITY ASSESSMENT: May not be technically feasible due to heat transfer requirements.
12. DU Metal Cask Integrated with Disposal Package

OPTION DESCRIPTION: This option would integrate the disposal package into the design of the DU Metal dry fuel storage cask. The disposal package would provide the interior steel shell now made up from steel in Option 2. The DU Metal would be sized to provide the added shielding. An external steel shell would provide the needed structural surface and attachment points for handling.

OPTION ADVANTAGES:

- System minimizes weight.
- Minimum diameter cask reduces disposal footprint.
- Simplifies heat removal from the fuel compared to Option 11.
- Simplifies fuel handling at repository and reduces facility requirements.
- Could eliminate fuel canister. Mechanical support system could be inserted directly into cask interior.
- Reduces depleted uranium utilization compared to other options with DU Metal (lower unit cost).

OPTION DISADVANTAGES:

- More complex assembly at utility location.
- High cost storage cask.
- DOE must provide as alternative for utility use.
- Requires expensive disposal package early in life of program, thus, increasing net present value of cost.

VIABILITY ASSESSMENT: Yes.
13. Disposal of U₃O₈ directly into Repository Empty Buffer Drifts

OPTION DESCRIPTION: This option disposes of depleted uranium oxide directly in buffer drifts. The uranium oxide would be packaged in drums or boxes to facilitate handling. Such containers would have limited lifetime but could be assured to last for the duration of the emplacement period. Containment during this period would be needed to contain the radon dose from the large quantity of depleted uranium. For packaging efficiency, some drift modifications might be necessary to maximize the volume of packaged uranium oxide. As there is essentially no heat load from the depleted uranium, the containers can be packed as close as physically possible in the existing excavated drifts.

OPTION ADVANTAGES:

- May require the excavation of additional drift space.
- Minimizes the cost of disposal and eliminates the need for a new geologic disposal facility.
- Utilizes existing repository drift space and reduces backfill closure work for these drifts.

OPTION DISADVANTAGES:

- Precludes the use of these buffer drifts for other purposes.

VIABILITY ASSESSMENT: Yes
14. Disposal of UO$_2$ DUAGG$^1$ Directly into Repository Empty Buffer Drifts

OPTION DESCRIPTION: This option is similar to option 14 and differs only in the physical form of the depleted uranium disposed. This option disposes of depleted uranium oxide in an aggregate form—DUAGG Aggregate—directly in buffer drifts. The uranium oxide would be packaged in large heavy bags to facilitate handling. The DUAGG Aggregate will have a minimum volume compared to the unstabilized oxide. Packaged and compressed bulk oxide will have a density of about 2.8 g/cc. DUAGG Aggregate will have a particle density of about 8.5 g/cc and assuming a 60% packing density, the bulk density could be about 5.1 g/cc. Improved packaging efficiency might be obtained with a second size aggregate.

Thus, considering that no containers are required other than large bags and the density advantage of DUAGG Aggregate, essentially twice as much uranium could be disposed per unit volume of repository compared to bulk uranium oxide powder. The ceramic aggregate itself is expected to minimize radon release, however, there are no data to support this. For packaging efficiency, some drift modifications might be necessary but these are expected to be less than for bulk containerized oxide. As there is essentially no heat load from the depleted uranium, the bulk bags can be packed as close as physically possible in the existing excavated drifts.

OPTION ADVANTAGES:

- May require the excavation of additional drift space.
- Provides for final disposal in an environmentally excellent final waste form.
- Minimizes the cost of disposal and eliminates the need for a new geologic disposal facility.
- Utilizes existing repository drift space and reduces backfill closure work for these drifts.

OPTION DISADVANTAGES:

- Precludes the use of these buffer drifts for other purposes.

VIABILITY ASSESSMENT: Yes

$^1$ DUAGG is the trademark name applied to the depleted uranium aggregate used to make DUCRETE Concrete. Both DUAGG and DUCRETE are trademarks of Lockheed Martin Idaho Technologies Company.