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EXPANSION OF THE CAPABILITIES OF THE GA-4 LEGAL WEIGHT TRUCK SPENT FUEL SHIPPING CASK

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ABSTRACT

General Atomics (GA) has developed the Model GA-4 Legal Weight Truck Spent Fuel Cask, a high capacity cask for the transport of four PWR spent fuel assemblies, and obtained a Certificate of Compliance (CoC No. 9226) in 1998 from the US Nuclear Regulatory Commission (NRC). The currently authorized contents in this CoC however, are much more limiting than the actual capability of the GA-4 cask to transport spent PWR fuel assemblies. The purpose of this paper is to show how the authorized contents can be significantly expanded by additional analyses without any changes to the physical design of the package. Using burnup credit per ISG-8 Rev. 2, the authorized contents can be significantly expanded by increasing the maximum enrichment as the burnup increases. Use of burnup credit eliminates much of the criticality imposed limits on authorized package contents, but shielding still limits the use of the cask for the higher burnup, short cooled fuel. By downloading to two assemblies and using shielding inserts, even the high burnup fuel with reasonable cooling times can be transported

1. INTRODUCTION

Under contract to the US Department of Energy (DOE) Office of Civilian Radioactive Waste Management (OCRWM) [1], GA developed the Model GA-4 Legal Weight Truck (LWT) Spent Fuel Cask. The design used advanced concepts and technologies to maximize the payload, and thereby reduce shipments with resultant significant life cycle cost savings. The resulting design was a high capacity LWT cask for the shipment of four pressurized-water reactor (PWR) spent fuel assemblies, compared to one assembly in other licensed LWT casks. When the DOE contract was terminated in 1995, GA continued cask development and licensing activities with private funds. GA undertook the modification of the neutron shield design, continued the licensing process, and obtained a Certificate of Compliance No. 9226 (CoC) in 1998 [Package ID No. USA/9226/B(U)F-85]. A twin LWT cask, the GA-9, was also developed using the same design principles, as a high capacity cask for BWR spent nuclear fuel (9 assemblies). The license application for this cask was docketed by the NRC

The payload capabilities of the GA-4 for fuel types, enrichment and burnup are significantly greater than those reflected in the current CoC, and the cask is capable of safely transporting most of the PWR spent nuclear fuel (SNF) assemblies currently in pool storage at nuclear power plants across the United States. Since the demand for the GA-4 was not expected to materialize until the Monitored Geologic Repository (MGR) at Yucca Mountain, Nevada was ready to accept fuel, it was decided at that time to delay much of the fuel specific licensing – which would allow removal of the CoC restrictions on authorized contents - until later.

Changes to the CoC that will significantly expand the authorized contents of the cask are described in this paper. In doing so, no changes to the GA-4 cask system are anticipated. The changes planned for a revision to the CoC are detailed criticality analyses to incorporate actinide-only burnup credit allowances and shielding analyses to evaluate the effect of downloading with shielding inserts. These analyses are also expected to reduce the B-10 neutron poison enrichment requirement.

2. CASK DESIGN DESCRIPTION

The GA-4 is a LWT cask capable of carrying up to four PWR spent fuel assemblies. The cask design was selected to enhance the overall safety and efficiency of the nuclear waste transportation system. The objective was to design a cask that would maximize payload and minimize the number of shipments, thereby minimizing life-cycle costs and exposure to the public. GA accomplished this objective by contouring the cask body shape around the payload of four PWR fuel assemblies as can be seen in Figure 1. This uncommon shape of flat sides with rounded corners contributes to achieving the capacity of four assemblies by minimizing the cask weight to keep it within the legal weight truck limit. The weight efficient depleted uranium (DU) gamma shield is also shaped to fit the shape of the contents. The thickness of the DU was adjusted at the corners to reflect reduced flux at these locations compared to the cask sides. The overall package dimensions, including impact limiters, are 229-cm (90-in) diameter by 595-cm (234-in) long, with a maximum gross weight of 25,000 kg (55,000 lbs), including 3,022 kg (6,648 lbs) in cask contents

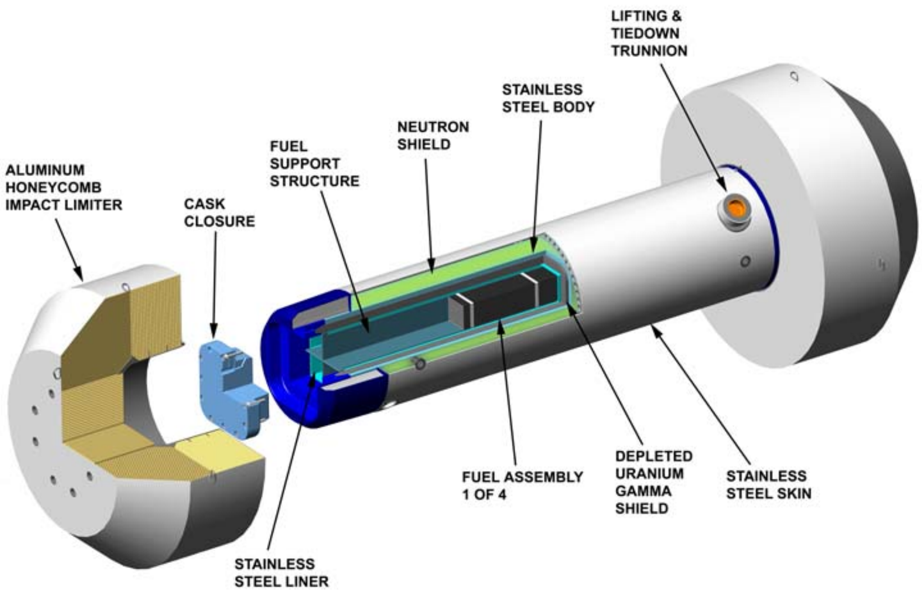

Figure 1. GA-4 Package Exploded VIew

The GA-4 cask uses a fixed fuel support structure to separate and support the fuel assemblies. It consists of welded XM-19 austenitic stainless steel (a nitrogen fortified alloy) plates with drilled holes to accept solid B_4C pellets. The use of B_4C permits a more compact array than would be possible using a matrix of boron and aluminium. The fuel support structure is welded to an XM-19 cavity liner. Outboard of the cavity liner is the DU and outboard of the DU is the cask body which is integral with the bottom head and welded to the cavity liner at the closure end.

The packaging includes the cask assembly and two impact limiter assemblies, each of which is attached to the cask with eight bolts. The aluminium honeycomb impact limiters were tested at ¼-scale to develop their design. They are more weight efficient than other impact limiter materials and also contributed to limiting the cask weight. The impact limiters transfer their crush loads to the cask body through radial ribs through the neutron shield. A cylindrical neutron shield shell surrounds the cask body and contains the neutron shield material.

The principal manufacturing processes and methods of forming, welding and machining to the close tolerances required were effectively demonstrated during the manufacture of a 1/2–scale model of the cask that accurately modelled all structural members (Figure 2). The model was fabricated and tested to confirm that the cask meets the 9-m drop and 1-m puncture USNRC regulatory requirements for accident conditions of transport (Title 10, Code of Federal Regulations, Part 71.73). Machining and assembly tolerances were also reduced to half-scale values for the 1/2–scale model, thereby demonstrating their feasibility for full-scale manufacturing.

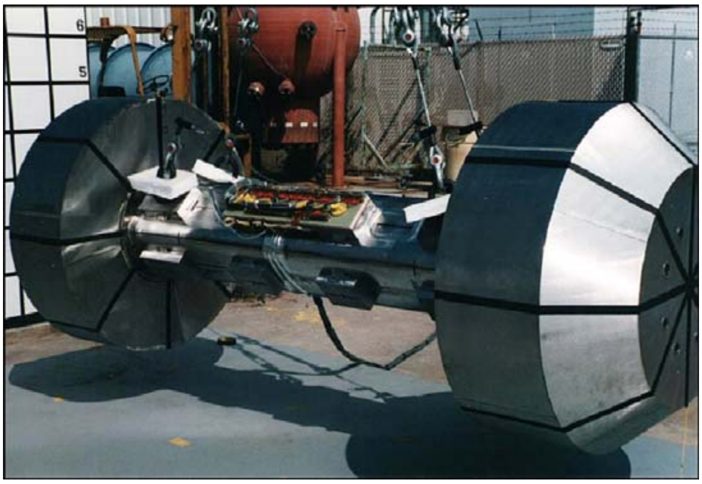

Figure 2 . Half-scale test model of GA-4 Cask

3. EXISTING LICENSE LIMITATIONS

In August, 1994, GA requested approval of its GA-4 LWT cask as a Type B(U)F package. The NRC issued the original five-year CoC in October, 1998 – and subsequently renewed in 2003 – with the following conditions:

The GA-4 is licensed to transport up to four intact PWR 14x14 and 15x15 spent fuel assembly arrays, with or without control rods or other nonfuel assembly hardware (NFAH). Spacers are used when shorter fuel elements are shipped. Shield inserts are optional when shipping fewer than four assemblies. The key restrictions on the cask loading in CoC 9226 currently are:

- Maximum initial enrichment for any assembly of 3.15 wt% U-235
- Maximum burnups of :
    a. 35 GWd/MTU with a minimum cooling time of 10 years.
    b. 45 GWd/MTU with a minimum cooling time of 15 years.
- Maximum allowable cask heat load of 2.468 kW.

These authorized contents are significantly more limiting than the actual cask payload capability. The cask as designed is capable of safely transporting most of the PWR spent nuclear fuel assemblies planned for transport to the MGR at Yucca Mountain, Nevada. Since the demand for use of the GA-4 cask system was not expected to materialize until the MGR was authorized to be constructed and receive fuel, further licensing efforts by GA – to remove the restrictions in the current CoC - were delayed until that time.

Under the current restrictions (which do not reflect actual cask capability), only about a quarter of the assemblies among the fuel types authorized (14x14 and 15x15 array assemblies) with greater than 5 years cooling could be shipped in the GA-4 cask. By applying burnup credit, amending the shielding analysis to allow downloaded configurations, and amending the structural and thermal analyses to allow for higher heat load assemblies, the GA-4 cask can be demonstrated to be capable of safely transporting nearly all the PWR inventory planned for shipment to the MGR without any structural modifications to the cask or ancillary systems. These proposed analyses are described below.

4. REVISED ANALYSES TO SUPPORT LICENSE AMENDMENT

Modifications to the contents currently authorized in the CoC can be supported by criticality and shielding analyses - without changes to the cask design - to significantly expand the number of spent fuel assemblies that can be safely shipped by legal weight truck using the GA-4 cask. These analyses are (a) actinide-only burnup credit to increase the allowable U-235 enrichment, and (b) shielding analyses to support downloading the payload for the higher burnup and shorter cooled fuel assemblies, and (c) thermal and structural analyses to allow higher power assemblies to be included in the authorized contents.

4.1 Burnup Credit. Criticality analyses were performed as part of the license application to establish the U-235 enrichment limits on the fuel assemblies that can be carried in the cask. At the time of the GA-4 license application submittal, the NRC was still developing its Interim Staff Guidance 8 (ISG-8) on the use of burnup credit for performing such analysis. Now that the NRC has completed its burnup credit guidance, the GA-4 license can easily be amended to follow ISG-8, Rev. 2 [2]. ISG-8 was developed after the NRC review of DOE's Actinide Only Burnup Credit Report [3]. General Atomics participated in the DOE burnup credit efforts by performing a sample analysis for the GA-4 contained in the appendix of Rev. 1 of the Topical Report [3]. Although there are some differences between the work done for Ref. 3 and the requirements of ISG-8, the loading curve generated is approximately correct. Analyses show that the enrichment limits for fresh fuel assuming loadings of three or two PWR assemblies are 3.7 and 4.5 wt% U-235 respectively. Using this information, loading curves for three and two assembly loadings of the GA-4 have been approximated. This approximation is done by assuming the shape of the curve generated for the four assembly loading is correct and then translating the curve to the right, to match the zero burnup points of 3.7 and 4.5 wt% U-235. The new enrichment limits after application of actinide-only burnup credit are shown in Figure 3, along with the burnup and cooling time limits related to shielding requirements.

4.2 Shielding Analysis. Shielding analyses performed as part of the license application determine the burnup limits that are applicable to the fuel assemblies that can be carried in the cask. In the GA-4 CoC, the burnup for the licensed payload of four PWR assemblies is limited to 35 or 45 GWd/MTU with 10 and 15 years cooling respectively. These loading constraints can be relaxed by the addition of more cooling times (17.5, 20, and 25 years cooled). Since shielding, not criticality, is the limiting factor for the higher burnup fuel assemblies, analyses have also been performed where the cask ships only two assemblies and uses shielding inserts in the other two fuel cells for shipping of fuel with increased burnup or less cooling time. The new limits for GA-4 payload capability based on downloading are also shown on Figure 3.

4.3 Thermal And Structural Analyses. The 0.617 kW decay heat limit imposed on the GA-4 at the time of the CoC application was a result of not needing a higher heat load given the criticality and shielding constraints. There is sufficient margin in the analyses to allow higher assembly average heat load of 0.75 kW for 4-assembly loading, 0.823 kW for 3-assembly loading, and 1.234 kW for a 2-assembly loading to allow loading and transportation of higher heat load PWR fuel assemblies in the GA-4 cask.

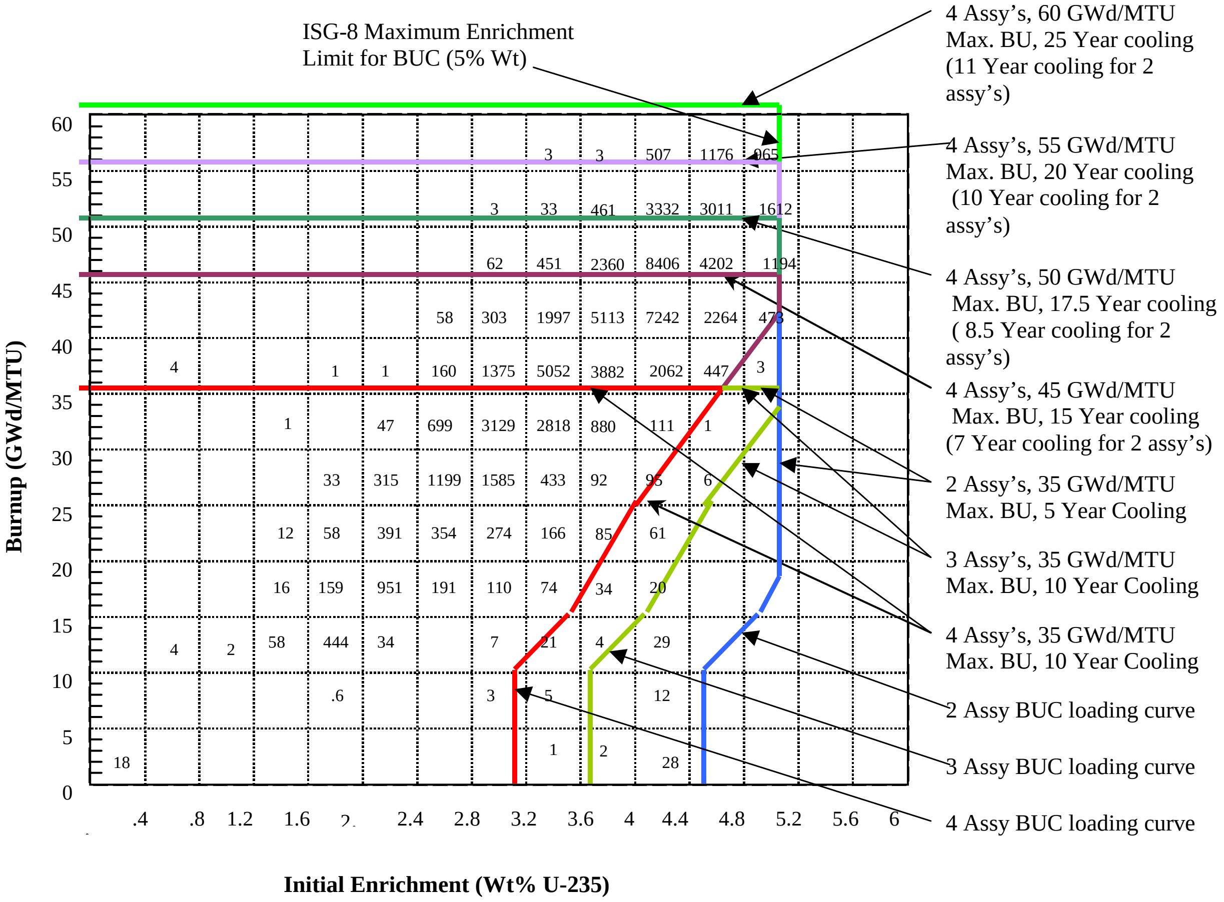

Figure 3. Estimated GA-4 loading curve with actinide-only burnup credit

5. CONCLUSIONS

The GA-4 LWT cask Certificate of Compliance (71-9226) can be modified using current NRC guidance in ISG-8, Rev 2, by performing additional criticality and shielding analysis to increase the authorized contents so that almost all of the PWR spent fuel inventory in utility pools can be shipped in the cask. This will allow shipment from truck-only utilities as well as the capability to ship spent fuel to the repository if rail transportation is not available for the MGR at Yucca Mountain.

Once the license amendments are implemented, the GA-4 cask can safely transport almost the entire PWR fuel inventory stored in the various reactor spent fuel pools in the United States (excluding Combustion Engineering 16x16 fuel assemblies - which could be placed in a GA 9 cask - and the South Texas Project fuel). If the fuel is only cooled 5 years (minimum cooling for "standard fuel" from 10CFR961.11) approximately 80% of the PWR SNF inventory planned for disposal at the Yucca Mountain MGR can be transported using the GA-4. For a seven year cooling period, the analyses show that greater than 90% of the inventory can be transported, and for fuel cooled to greater than 11 years, the entire PWR inventory (with the exceptions noted above) can be shipped. For the very few assemblies that require a reduced cask loading due to criticality constraints, it is expected that fission product burnup credit will eventually be approved, which will then allow a full loading.

6. REFERENCES

1. Work supported under U.S. Department of Energy, Office of Civilian Radioactive Waste Management, Contract No. DE-AC07-88ID12698.
2. *Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel In Transport and Storage Casks*, US Spent Fuel Project Office, US Nuclear Regulatory Commission, Interim Staff Guidance - 8, Revision 2, September 2002.
3. *Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages,* DOE/RW-0472 Rev.1, Office of Civilian Radioactive Waste Management, US Department of Energy, May 1997 (Rev. 2, September 1998)