GA-A--20772

DE93 001947

BURNUP CREDIT APPLICATIONS IN A HIGH-CAPACITY TRUCK CASK

BY
JACK K. BOSHOVEN

This is a preprint of a paper to be presented at
PATRAM '92, September 13-18, 1992, in Yokohama, Japan

Work Supported by
Department of Energy
Contract DE-AC07-88ID12698

GENERAL ATOMICS PROJECT 3462
SEPTEMBER 1992

GENERAL ATOMICS

Burnup Credit Applications in a High-Capacity Truck Cask*

Jack K. Boshoven

General Atomics, San Diego, California

ABSTRACT

General Atomics (GA) has designed two legal weight truck (LWT) casks, the GA-4 and GA-9, to carry four pressurized-water-reactor (PWR) and nine boiling-water-reactor (BWR) fuel assemblies, respectively. GA plans to submit applications for certification to the U.S. Nuclear Regulatory Commission (NRC) for the two casks in mid-1993. GA will include burnup credit analysis in the Safety Analysis Report for Packaging (SARP) for the GA-4 Cask. By including burnup credit in the criticality safety analysis for PWR fuels with initial enrichments above 3% U-235, public and occupation risks are reduced and cost savings are realized. The GA approach to burnup credit analysis incorporates the information produced in the U.S. Department of Energy Burnup Credit Program. This paper describes the application of burnup credit to the criticality control design of the GA-4 Cask.

BACKGROUND

The light-water-reactor (LWR) spent fuel shipping casks currently certified by the NRC are designed to transport spent fuel which has been out of the reactor for a minimum of 150 days. Cask capacities are limited by thermal and shielding requirements for the 150-day-old fuel. GA is designing the next generation of spent fuel casks to carry 10-year out-of-reactor fuel. By using two dedicated casks and because of the reduced heat generation and reduced neutron and gamma sources, GA has been able to increase the LWT cask payloads from one PWR assembly to four and from two BWR assemblies to nine. The increased capacity of PWR fuel in the GA-4 Cask, however, makes criticality control the new limiting factor in the design.

In the past, criticality safety analyses for LWR spent fuel shipping casks have assumed the cask to be loaded with fresh fuel without burnable poisons. Ignoring the burnable poison and not taking credit for the depletion of U-235 and the buildup of actinides and fission products due to fuel burnup is very conservative since fresh fuel without poisons is the most reactive state possible for criticality purposes. The GA-9 Cask meets the

* Work is supported by the U.S. Department of Energy, Office of Civilian Radioactive Waste Management, under DOE Field Office, Idaho, Contract No. DE-AC07-88ID12698.

criticality safety requirements using the fresh fuel assumption. The GA-4 meets the criticality safety requirements using the fresh fuel assumption for fuel assemblies having initial enrichments less than 3 wt% U-235. For higher enrichments, the GA-4 requires partial credit for the depletion of U-235 and the buildup of actinides and fission products that occur during burnup (i.e., burnup credit).

If the fresh fuel assumption is made for the criticality safety analysis, the capacity of the GA-4 Cask is reduced by 25% to three spent fuel assemblies. By taking partial credit for the burnup of the spent fuel in the criticality safety analysis with initial enrichments over 3.0 wt% U-235, the capacity of the GA-4 Cask is increased to four assemblies. This increase in cask capacity results in significant reductions in public and occupational risks and cost savings (Sanders, 1989).

GA-4 Design

GA designed the GA-4 Cask to meet Fissile Class I Package requirements as defined in 10CFR71 (NRC, 1983). Figure 1 shows the cross section and dimensions of the GA-4 Cask design. Each layer of structure and shielding closely fits around the fuel cavity to minimize weight. The shaped design with optimized shielding in the rounded corners saves about 3200 kg (7000 lb) over a round cask with the same capacity. The neutron shield material is also tapered at the ends to save weight. GA further reduced the GA-4 Cask weight by placing the most dense materials toward the inside of the cask. The first layer outside the fuel cavity is depleted uranium (DU) gamma shielding. The next layer is the XM-19 austenitic stainless steel pressure vessel wall. Outside the stainless steel structure is the solid borated polypropylene neutron shield material. The entire cask is encased in a smooth stainless steel skin to facilitate decontamination.

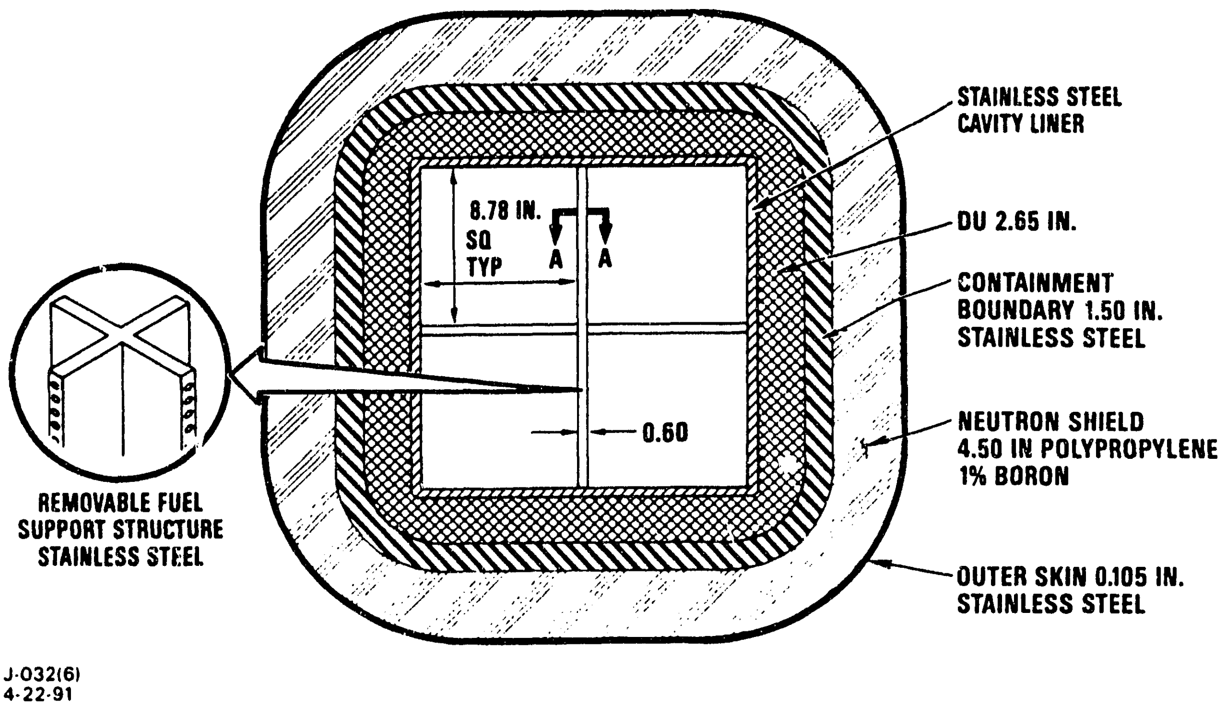


Fig. 1 GA-4 Cask Cross Section

The fuel support structure (FSS) inside the fuel cavity provides criticality control. The FSS has four panels, each of which encases solid boron carbide (B_4C) pellets. A continuous series of holes in each panel, located at a right angle to the FSS axis, provides cavities for the B_4C. GA chose B_4C because it maximizes the boron loading for criticality control. Boron carbide is a solid compound used in control rods because of its effectiveness and ability to be sintered to densities of 2.47 g/cm^3 or more.

Criticality Analysis With Burnup Credit

GA uses the Westinghouse 17 x 17 standard (STD) fuel assembly as the reference fuel type for the criticality analysis. The Westinghouse 17 x 17 STD is often used for shipping and storage cask criticality safety analysis because of the large quantity in the existing inventory. Since the Westinghouse 17 x 17 STD assembly is not always the most reactive fuel for different levels of burnup, GA adds a bias as a function of burnup to the analysis to take this into account.

In accordance with 10CFR71, the worst-case cask configuration with optimum moderation must be subcritical. As a matter of practice, this is accomplished by maintaining $k_{eff} \leq 0.95$ including all biases and uncertainties. The worst case configuration for the GA-4 Cask is an undamaged cask filled with water. The following conservative assumptions are used for calculating k_{eff}:

- Minimum fuel assembly pitch within the cask
- Omission of grid plates, spacers and hardware in the fuel assembly
- No burnable poisons
- Water density at 1.0 g/cm^3 and at 20°C (293 K)
- The polypropylene is modeled without the boron.
- B_4C density of pellets assumed to be 93% theoretical density (supplier will guarantee 98% theoretical density).

The fuel assembly data required for the criticality analysis include the physical dimensions and isotopic data for burned fuel. The physical data for the Westinghouse 17 x 17 STD fuel assembly are available in *Characteristics of Spent Fuel, High Level Waste and Other Radioactive Wastes Which May Require Long-Term Isolation* (DOE, 1987). Table 1 reproduces the pertinent data used in the criticality analysis.

GA used the SAS2H module (ORIGEN-s) of the SCALE-4 system of codes (ORNL, 1990) to determine the composition of the spent fuel for the burnup credit analysis. Table 2 lists the nuclides accounted for in the analyses. These isotopes represent all of the fissionable nuclides and only those fission products which are responsible for 80 percent of the neutron absorptions for all fission products in 5-year out-of-reactor fuel. GA chose a cooling time of 5-years to provide the flexibility of shipping 5-year cooled fuel in the cask.

TABLE 1
KEY PARAMETERS FOR CRITICALITY ANALYSIS

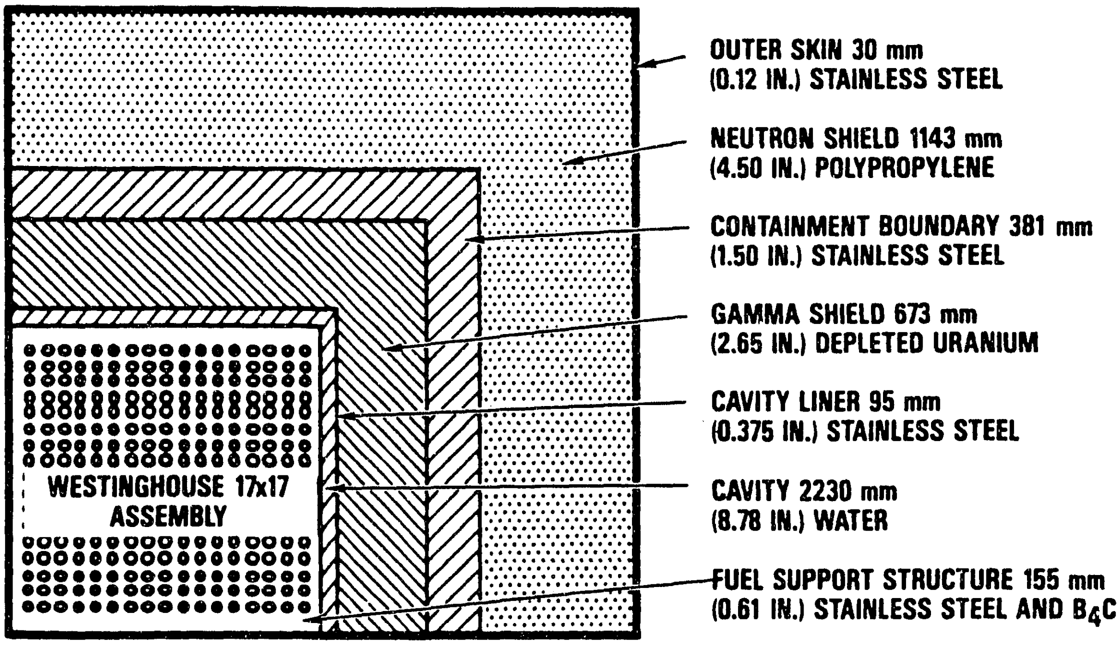
Description	Parameter
Fuel Support Structure Poison Material	B_4C rod
Minimum B4C Rod Diameter (in.) Small Pellets Large Pellets	 0.278 0.426
B4C Rod Pitch (in.)	0.5
Maximum Fuel Cavity Width (in.)	8.796
Fuel Type	W 17 x 17 STD
Fuel Assembly Pitch	Minimum
Number of Fuel Rods Number of Water Holes	264 25
Fuel Rod Pitch (in.)	0.496
Fuel o.d. (in.)	0.329
Cladding Thickness (in.)	0.0225
$U0_2$ Smear Density (g/cm3)	8.77
Cooling Time (years)	5.0

TABLE 2
SPENT FUEL ISOTOPES INCLUDED IN CRITICALITY ANALYSIS

U-234	Pu-239	Tc-99	Sm-151	Mo-95
U-235	Pu-240	Sm-149	Sm-152	Sm-150
U-236	Pu-241	Rh-103	Eu-153	Cs-133
U-238	Pu-242	Nd-143	Nd-145	Cs-135
Pu-238	Am-241	Gd-155	Sm-147	O

GA used the CSAS25 control module (KENO V.a) of SCALE-4 to determine the effective neutron multiplication (k_{eff}) of the cask and fuel. The nuclear data used for the criticality calculations is the 27-group cross section library 27-BURNUPLIB built into SCALE-4. This broad-group library was developed by Oak Ridge National Laboratory (ORNL) specifically for criticality analysis of a wide variety of thermal systems based on ENDF/B/IV data.

GA modeled the cask using a full height quarter section with reflective boundary conditions on the x and y axes. The model explicitly models the B_4C pellets at minimum length and diameter in the center of the maximum diameter hole. Figure 2 illustrates the cask model used for the criticality analysis. A square cross section was used because of the geometry limitations in the KENO V.a. GA used the standard material data library in SCALE-4 for the cask materials in the model. The Westinghouse 17 x 17 STD fuel assembly is also explicitly modeled as a 17 x 17 array comprising 264 fuel rods, including fuel and cladding and 25 water holes. The fuel is modeled with



K-632(1)
8-6-92

Fig. 2 KENO V.a Model of the GA-4 Cask

uniform axial burnup except for the cases used to determine the effect of the axial burnup profile and k_{eff}.

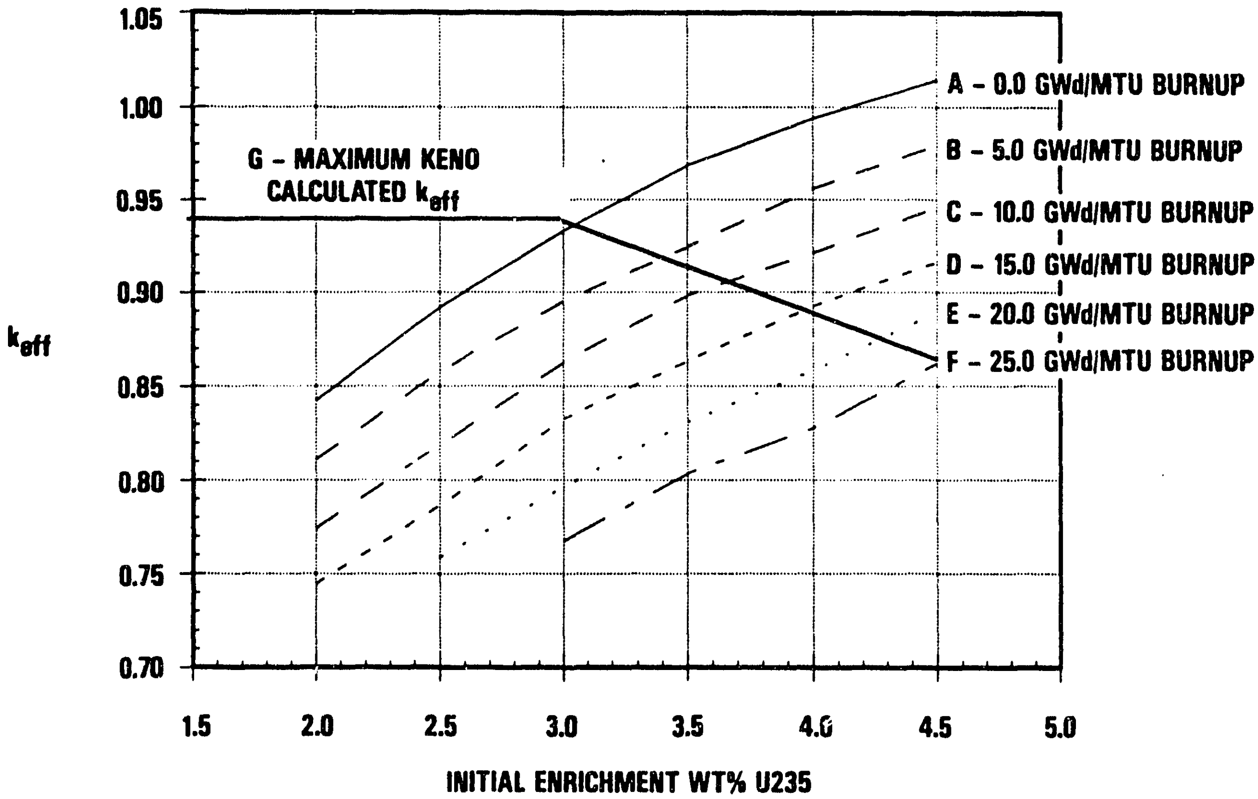
Curves A through D in Fig. 3 graphically present the results of the criticality calculations. These curves represent the k_{eff} as produced with KENO V.a without bias and uncertainty added. Curve E, which shows the minimum burnup required for $k_{eff} \leq 0.95$, includes all biases and uncertainties.

Bias and Uncertainties Associated With Burnup Credit

The fresh fuel assumption traditionally used in cask criticality analysis makes the determination of the biases and uncertainties straightforward. The criticality code used in the analysis is benchmarked against a set of critical experiments which are representative of the cask materials and configuration to determine the code bias. The uncertainties can be quantified with simple sensitivity analysis which depends only on a few variables. For burnup credit analysis, a second code used for isotopic generation must be benchmarked with its associated bias. In addition to this new bias many other burnup dependent uncertainties must be taken into account in the analysis. Table 3 lists the biases and uncertainties taken into account in the GA analysis. Curve E on Fig. 3 graphically displays how these biases and uncertainties are taken into account for determining the minimum burnup required to assure criticality safety for the GA-4 Cask. The difference in k_{eff} between curve E and 0.95 represents the increase in uncertainty as a function of burnup. Although the uncertainties due to burnup credit may be able to be reduced in the future, they can never be reduced to the level of those for the fresh fuel assumption; furthermore, the uncertainties tend to increase with burnup.

TABLE 3
BURNUP CREDIT BIASES AND UNCERTAINTIES

Bias/Uncertainty	Δk_{eff} at 25 GWd/MTU Burnup
KENO V.a BIAS	0.02 (0.0078 fresh fuel)
ORIGEN-s BIAS (fuel isotopic content uncertainty)	0.03
End Effects (axial burnup profile)	0.02
Operating History	0.01
Most Reactive Fuel	0.004
Total	0.09



K-643(1)
8-18-92

Fig. 3 Criticality Safety Design Curve

Conclusions

The use of burnup credit in the criticality safety analysis of the GA-4 Cask increases the cask's capacity from three spent fuel assemblies to four, resulting in reduced public and occupational risk and reduced life cycle costs. GA's criticality calculations for burnup credit, including the associated uncertainties and analytical bias, establish the minimum burnup required as a function of initial enrichment to maintain $k_{eff} \leq 0.95$ under any conceivable condition. The minimum burnup requirement as a function of initial enrichment has been determined to be 15,000 MWd/MTU for 3.5 wt% U-235 fuel, 20,000 MWd/MTU for 4.0 wt% U-235 fuel and 25,000 MWd/MTU for 4.5 wt% U-235 fuel. The minimum burnup requirement as a function of enrichment is well below the typical burnup levels seen in the current and projected spent fuel inventory.

REFERENCES

Characteristics of Spent Fuel, High-Level Waste and Other Radioactive Wastes Which May Require Long-Term Isolation, DOE/RW-0184, (December 1987).

Computer Code Collection *SCALE-4, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,* Vol. 1-3, CCC-545, Oak Ridge National Laboratory, (February 1990).

Sanders, T. L., Editor, *Proceedings of a Workshop on the Use of Burnup Credit in Spent Fuel Transport Casks,* SAND89-0018, (October 1989).

U.S. Nuclear Regulatory Commission, *Packaging and Transportation of Radioactive Materials,* Code of Federal Regulations, Title 10, Part 71, Washington, DC (1983).

END

DATE FILMED

2/12/93