Shield Design Methodology and Validation of Analysis Methods for Transport Casks

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INTRODUCTION

British Nuclear Fuels plc has for some years been engaged in the transportation of irradiated nuclear fuel from LWR sites in Japan and Europe, to the Sellafield reprocessing complex in the United Kingdom. This business has used casks designed and licenced by the Company. More recently, the Company has also designed a cask to return vitrified waste to the Country of origin for the reprocessed fuel. Consequently, the Company has established a considerable expertise in the design and licencing of transport casks for the transport of fuel and radioactive waste.

The evaluation of a transport cask design requires that a complex iteration is undertaken to optimise, amongst other things, cask size and weight, heat transfer properties, shield design and, of course, cost. The objectives of this paper are:

(i) to outline the shielding design methodology used by BNFL for the design of its casks;

(ii) to describe work commissioned by BNFL to experimentally validate the neutron and gamma ray shielding properties of a cask.

TYPICAL TRANSPORT PACKAGE CONFIGURATION

The international transport of irradiated nuclear fuel must comply with maximum radiation levels stipulated by the IAEA in the "Regulations for the Safe Transport of Radioactive Materials". This is achieved by the characterisation of the fuel to be transported from the reactor utilities and subsequently the calculation of the necessary shielding thickness.

Individual fuel elements for transport are held within a multi-element bottle. This bottle accepts the irradiated fuel elements in an array of square channels arranged and supported within the bottle shell. Sheets containing a dispersion of boron carbide and aluminium in a sandwich of aluminium are inserted within the array and surrounding each fuel element to provide an adequate margin to critical. The bottle is closed by a lid.

The multi-element bottle is transported within a water filled cask. A typical cask comprises a 90mm thick cylindrical mild steel shell, with a 300mm thick mild steel lid and base. It contains a 160mm thick lead liner which extends for the internal cavity length of the cask. The cavity diameter is about 1.3m and the cavity length about 5m. Cooling of the cask is achieved by circumferential fins extending along virtually the whole cask body. An example of such a cask and multi-element bottle is shown in Figure 1.

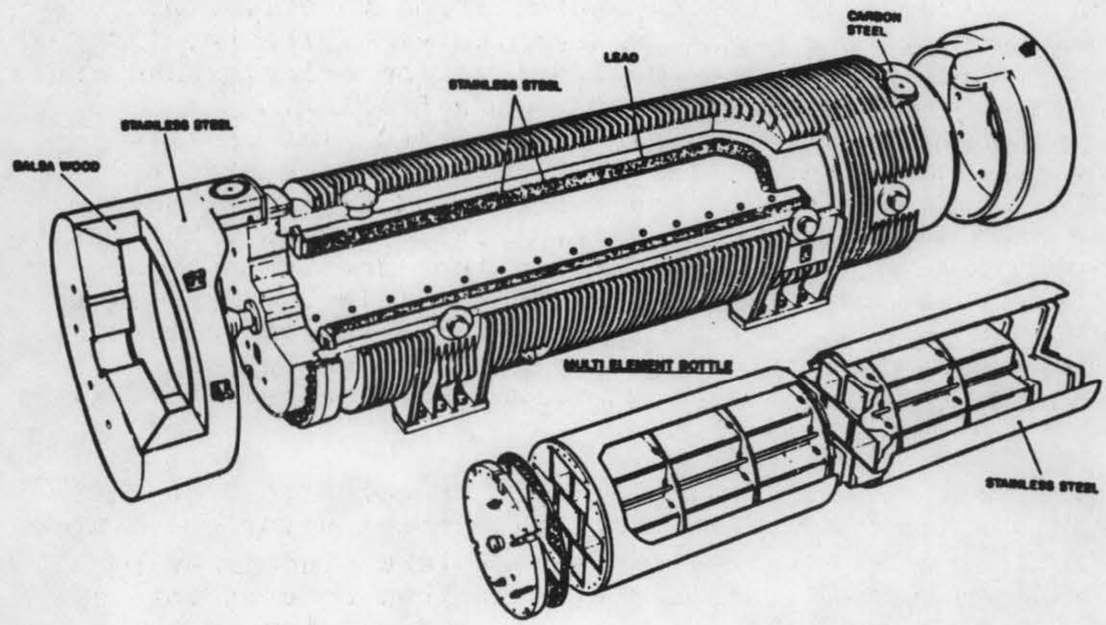

FIG. 1 EXCELLOX IV IRRADIATED FUEL TRANSPORT FLASK

The principal gamma ray shield comprises the mild steel lid and base of the cask and also the mild steel shell and lead liner along the length of the cask. Neutron shielding is largely accomplished by the water present in both the multi-element bottle and also the cask.

SHIELD DESIGN METHODOLOGY

Overview

Shielding calculations require the characterisation of the fuel to be transported within the cask. For a Company such as BNFL who service many reactor utilities this entails establishing, both the range of fuel designs to be transported and then also the operating parameters of the fuel, including physical data such as length or number of pins, composition data such as Co59 content of the steel and also nuclear data such as initial enrichment , burnup, rating etc. To achieve an optimum cask design, these data must be sufficiently pessimistic to permit the cask to operate without unduly restricting the payload and hence economic viability of the transport.

Principle Methods

Within the United Kingdom, an independent, set of methods and data have been developed which are all fully validated and form a comprehensive shielding assessment route.

The assessment procedure uses the inventory code FISPIN which performs a three group calculation to determine fission product and actinide inventories of the previously characterised irradiated fuel. Activation calculations are again performed in three groups using FISPIN in conjunction with a previously established flux profile along the fuel element. The isotopic data generated by FISPIN form the basis of the neutron and gamma ray source terms used in subsequent shielding calculations. The contribution to the neutron source term due to neutrons emitted by the capture of an alpha particle by the nucleus of a light element is also included. The source spectra for neutrons are typically modelled in 27 energy groups and the source spectra for gamma ray photons are typically modelled in 15 energy group.

Sub critical neutron multiplication is calculated in three dimensions for the irradiated fuel element using the MONK criticality code. The neutron source term, increased by the neutron multiplication factor is then entered into the MCBEND code. This code uses an enhanced version of the three dimensional combinatorial geometry (CG) package to generate a detailed computer model of the cask.

A notable feature of the calculational route adopted by BNFL is the use of the code FENDER to solve the adjoint form of the diffusion equation and hence generate an accurate multigroup acceleration function for the MCBEND code.

Primary gamma ray calculations are normally performed using the RANKERN code, which is a three dimensional, point kernel, stochastic integration code. This code again uses the CG package, and calculates the dose rate contribution due to the scattered flux through the use of Capo buildup factors. In cases where the buildup factor approximation is not acceptable, a more rigorous gamma ray calculation would be performed using MCBEND. In all cases, the study of secondary gamma rays would be performed using MCBEND.

Cross check methods

Due to the international nature of the BNFL reprocessing business, much experience has been developed in the use of the American codes ORIGEN, QAD-CG, ANISN and DOT. In general little difference is found between ORIGEN and FISPIN or QAD-CG and RANKERN. More substantial discrepancies occur between ANISN and DOT when compared with MCBEND; however given the geometrical limitations of these codes when compared to MCBEND these differences are acceptable for the purpose of cross check calculations.

To develop an independent and novel means of checking the MCBEND calculations BNFL and University of London have jointly developed the code FELTRAN, which is described in "Finite element computer code in solving the multigroup and anisotropic neutron transport equation",J.G.Issa, 1988. This code solves, in two spatial dimensions, the second order, even parity form of the Boltzmann transport equation using the method of finite elements in conjunction with an expansion of the angular flux in spherical harmonics. This code is at present being extended to three spatial dimensions.

Nuclear data

MONK and MCBEND both use an 8000 energy point representation of cross section, derived primarily from the United Kingdom Nuclear Data Library (UKNDL). This is the principle source of data used for neutronics calculations. However both MONK and MCBEND can also use multigroup data.

In the event of a discrepancy, at the cross check calculation stage between say, MCBEND using point data and

DOT using multigroup data the ability to load the same multigroup data into MCBEND gives the ability to identify if the source of the discrepancy lies in the code or data.

VALIDATION OF CALCULATIONS USING MCBEND

To demonstrate the confidence in the BNFL calculational route a validation exercise was undertaken using eight accurately calibrated Cf252 sources, located within a cask intended for transporting strong neutron sources. At the time of the experiment the total source was measured by the National Physics Laboratory to be $2.72 \times 10^7 \pm 0.52\%$ neutrons/second.

The carrying capacity of the cask is provided by four cylindrical channels each set in a block of ferralium. These blocks are mounted on the sides of a square inner frame of stainless steel and are connected by copper plates, two per block, to the mild steel cylinder which forms the outer casing. The remaining space within the cask body is filled with wood, American White Oak, to give the required neutron attenuation. Each channel is provided with a ferralium cap and the lid of the cask fits over these, being secured by twelve bolts arranged in four groups of three at 90 degrees intervals around the rim. The lid itself is also a mild steel frame filled with wood and with four cut-outs to accommodate the retaining bolts.

Measurements of the neutron dose rate were made with a Mk7 NRM Dose-Equivalent meter. The latter is a spherical counter with an outer radius of 10.45cm so that the measurements near the surface of the cask were made with its centre at a distance of 11.0cm. The accuracy of the dose rates returned with this instrument is estimated to be ± 10%. The gamma ray dose rates were measured with a survey instrument Type 1907B. Checks on the accuracy of these measurements were made by comparison with doses determined using LiF thermoluminescent detectors at ten locations where the levels were sufficiently high to give meaningful readings after 65 hours exposure. These comparisons indicated an accuracy of ± 20% for the dosimeter readings. The arrangement of the cask is shown in Figure 2.

Surveys of the dose rates were made at points on the centre plane of the cask extending to a distance of one metre. For these measurements the cask was mounted on a cube of concrete of side 61cm so that it stood with its axis vertical and its lid uppermost.

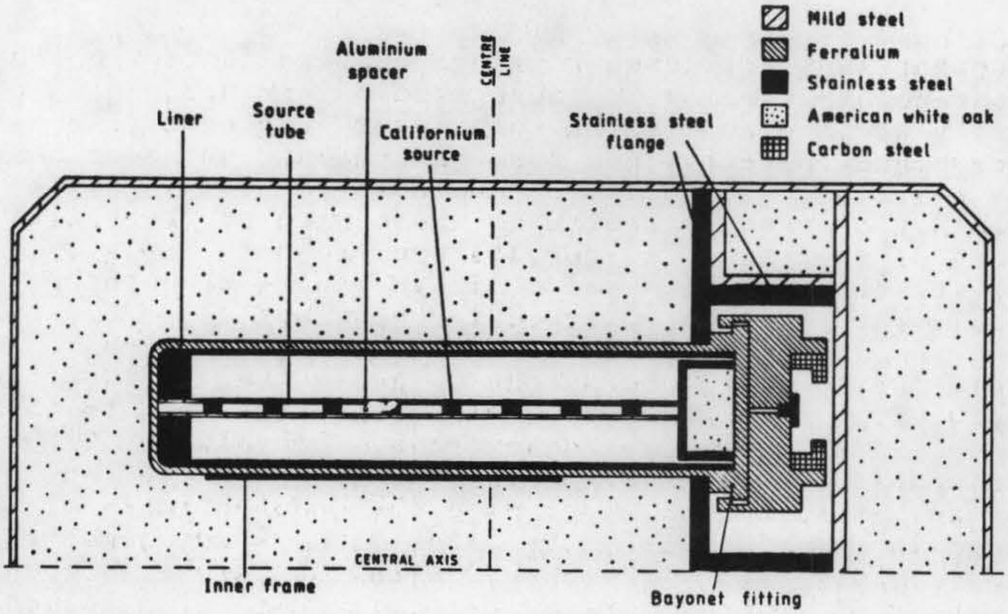

FIG. 2 VERTICAL CROSS-SECTION OF THE FLASK PASSING THROUGH THE CENTRE OF THE SOURCE CAVITY ($\theta = 0°$)

The attenuation of neutron dose rate with distance from the cask as given in Table 1 for points on the centre plane shows a fall in the ratio of calculated to experimental values (C/E) from 1.19 at 10cm to 0.85 at 100cm. An increasing underestimate in the falloff of dose rate with distance is to be expected with the low values of the calculated neutron dose rates leaking from the radial surface in the region of the cask lid. This would have a larger effect as the contributions from the axial extremes of the radial surface become more significant at the remote dose-points.

TABLE 1

Neutron dose rates on the centre plane of the cask

Distance from surface	Neutron dose rate (μSv/h)		C/E
	Calculated	Measured	
10	516.2 ± 7%	435.0	1.19
20	344.5 ± 7%	308.5	1.12
40	214.0 ± 8%	181.5	1.18
60	129.0 ± 9%	122.5	1.05
80	88.8 ± 10%	94.3	0.94
100	59.2 ± 10%	69.4	0.85

The comparisons for gamma rays are carried out for a range of dose-points similar to those studied for neutrons. The quantity of interest is the total gamma ray dose rate with the combined contributions from the primary and secondary gamma ray sources. The latter dominate the total giving 60% to 80% of the dose rates at the centre of the radial surface of the cask. The results are shown in Table 2 where the trends in the value of C/E can be seen to be very similar to those observed for neutrons.

The similarity of the behaviour suggests that it is errors in predicting the neutron distribution which lead to those observed in the gamma ray dose rates. The value of C/E again shows a decrease with distance from the cask and this would also be explained with the underestimate of leakage in the lid region.The values of C/E for gamma rays range from 0.93 to 0.72 compared with the values between 1.19 and 0.85 found for neutrons. This level of agreement is considered to be satisfactory for a cask where the main shielding material is wood with a specified minimum moisture content which may be exceeded.

TABLE 2

Gamma ray dose rates on the centre plane of the cask

Distance from surface (cm)	Gamma ray dose rate (μSv/h)		C/E
	Calculated	Measured	
2	72.6 ± 8%	78.3	0.93
10	51.9 ± 9%	55.8	0.93
20	36.8 ± 9%	39.6	0.93
40	21.0 ± 8%	24.3	0.86
60	13.3 ± 8%	16.0	0.83
80	9.9 ± 9%	11.8	0.84
100	7.1 ± 10%	9.8	0.72

CONCLUSIONS

The cask shielding design capability of BNFL has been outlined and shown by means of a validation experiment to be accurate. The calculation route is based upon methods and data developed within the United Kingdom.

It is concluded that the shield design methodology adopted by BNFL for its transport cask assessments is comprehensive, independent and validated. This forms a sound basis for future BNFL cask designs or for BNFL undertaking an independent review of cask designs by other organisations.

ACKNOWLEDGEMENT

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