High Polymer-Based Composites for Spent Nuclear Fuel Disposal Containers

H.W. BONIN and V.T. BUI

Department of Chemistry and Chemical Engineering
Royal Military College of Canada / Collège militaire royal du Canada
P.O. Box 17000; Station FORCES
Kingston, Ontario Canada K7K 7B4

ABSTRACT

The concept proposed by Atomic Energy of Canada Limited for the deep underground disposal of spent nuclear fuel foresees the use of metals or alloys such as copper and titanium for the fabrication of the storage container. The choice of these materials is based mostly on their good resistance to adverse environments and corrosion in particular, such that they are believed to be capable of keeping their integrity over the many centuries the radioactive materials have to be kept isolated from the biosphere. The authors have started a research project aimed at proposing alternative materials for the fabrication of the containers such as high polymer-based composites known for their excellent mechanical and chemical resistance properties. This paper will present the results obtained so far on the radiation damage and the heat transfer aspects of this special application.

INTRODUCTION

The spent nuclear fuel from CANDU reactors makes up the major part of the high level radioactive level waste generated in Canada. The disposal of such hazardous material is currently the object of intense and passionate debate. Following a multi-year research and development programme, Atomic Energy of Canada Limited is proposing (1-6) the indefinite, non-retrievable, storage in deep underground vaults within the plutonic granite rock of the Canadian Shield. Based on a conservative engineering approach, the concept involves some aspects of advanced technology, such as for the materials selected for the fabrication of the storage container which are titanium-based alloys, known for their good resistance to corrosion. However, new advanced materials have been developed recently, such as several types of high polymer-based composites, which display comparable properties to the alloys, when not superior. These composite materials offer excellent resistance to corrosion in addition to their light weight and superior mechanical strength (7-20). While this research considers the disposal of spent CANDU fuel bundles, the application to other types of radioactive waste is straightforward. A typical container, as shown in Figure 1 would store 72 CANDU bundles and, once fully loaded and hermetically sealed, it would be placed in a cavity dug in the floor of an underground disposal vault or in a chamber carved in the wall of the vault located at some 500-1000 m deep. The vault

would be connected to other vaults by a network of tunnels bored deep in a plutonic rock formation within the Canadian Shield. The choice of a such rock formation comes from the fact that plutons are homogeneous granite monoliths that have "survived" intact (or with little damage) the geologic movements of the earth's crust that have occurred in the last billions of years.

Inside its chamber in the disposal vault, the container would be surrounded by buffer materials typically containing clay and intended to limit the corrosion rate of the container material and the rate of dissolution of the waste form should ground water manage to seep into the container. Furthermore, the buffer material would prevent or impede the movement of the contaminants in the event that they are released from the container. The chamber, the disposal vault and, in turn, the access tunnels would then be filled with backfill material made of cement, clay or concrete. The function of this is again to prevent or retard the movement of any contamination escaping through the buffer layer, and to firmly secure in place the containers and the surrounding buffers. Once all the storage sites within the disposal facility are occupied, all of the remaining tunnels and access shafts and other boreholes would be definitely sealed with clay-based or cement-based materials intended not only to prevent or retard the migration of radioactive contaminants toward the surface, but also to prevent access to the radioactive waste by humans or animals. The concept is therefore based on the multi-barrier approach adopted by AECL to protect the biosphere from the radioactive contamination: nuclear fuel matrix and fuel cladding, filler material (glass beads are proposed by AECL), container wall, buffer material, backfill material and 500-1000 m thick plutonic granite layer.

MATERIALS PROPOSED FOR THE CONTAINER WALLS AND FILLERS

This research aims at evaluating the feasibility of using high polymer composite materials for the fabrication of the storage container. The AECL design for the container was retained as for its dimensions and capacity: 2.246 m height, 0.633 m diameter and 6.35×10^{-3} m thickness, for a 72 CANDU fuel bundle capacity. Four polymeric composites were examined: polystyrene (PS), polymethyl methacrylate (PMMA), Devcon #10210 Epoxy, and polyetheretherketone (PEEK). For all these materials, the reinforcing material was boron fibres in either a 50% (mass) or a 70% (mass) mixture. Several characteristics of these composite materials such as mechanical strength, resistance to corrosion, heat transfer properties and resistance to radiation were investigated and the performance of these materials was analysed and compared with the titanium alloy proposed by AECL and with copper (6) (proposed by the Chemical Institute of Canada (21)).

Polystyrene (PS) is an amorphous, clear, brittle polymer that can be injection molded extruded, filament wound and blown to produce containers. Below its 100°C glass transition temperature, it possesses considerable mechanical strength and a high resistance to water and several other chemical agents. Polymethyl methacrylate (PMMA) is also used in extrusions, molding and as protective coatings and adhesives. It displays an excellent resistance to corrosion and only concentrated acids or alkalies can attack this polymer. Its glass transition temperature is only 105°C and may retain its glass state properties if used

as a radwaste container. Again, the mechanical strength of this polymer is quite high and it was included in this study, like the PS, because it was readily available and for knowing more on its behaviour under radiations.

Epoxies have been investigated at RMC-CMR since several years (22-24) and they have displayed good resistance to radiations, even to the point of actually strengthening when subjected to moderate doses of neutrons and gammas. These high polymer compounds are thermosetting resins that cure when heat is applied (often resulting from chemical reactions from the hardening agent) and form highly cross-linked, insoluble and infusible matrix resins. They display very high chemical stability and resistance to corrosive agents in addition to outstanding adhesive and mechanical strength. There is a wide variety of epoxies available for specific applications, and using a wide range of curing agents such that the setting time varies from a few minutes to 24 hours or more. The Devcon 10210 epoxy chosen for this work is a general purpose variety with a 24-hour curing time and is often used in the fabrication of composite materials.

PEEK (Polyetheretherketone) is a semicrystalline material with superior characteristics such as resistance to high temperatures, radiations and corrosion (25). It displays excellent mechanical properties such as providing high wear resistance and may well be used in applications in high temperature environments (about 300°C). As demonstrated in another work (26, 27), its resistance to radiations is much higher than for most of the polymers. The glass transition temperature is 143°C, which does not represent a problem for application in the container fabrication. The tensile strength of PEEK exceeds that of most of the other engineering plastics and, with reinforcement, tensile strengths over 100 MPa can be obtained and maintained at temperature above 300°C.

Boron is selected as the reinforcement fibre mostly because of its excellent mechanical and chemical properties at elevated temperatures (28). The high absorption cross section for boron for thermal neutrons (759 b) is also a desirable asset of this material, although the intensity of the thermal neutron flux at the position of the container wall would be very small making the neutron absorption in these fibres negligible indeed. It is rather the excellent properties of resistance to corrosion and outstanding compressive strength, plus the ever increasing uses of this material in the fabrication of composites which justify this choice.

In addition to evaluating glass beads suggested in the AECL proposal for the container filler material, this work also looks at thorium dioxide (ThO_2) for this filler. This choice is based on the excellent physical and chemical properties of this ceramic, such as excellent resistance to corrosion by air and water and very high melting point (3050°C) (29). Although mildly radioactive itself, thorium-232 has a very long half-life (1.39×10^{10} years), longer than the two natural uranium isotopes and its presence even in more concentrated form wouldn't add much to the natural radioactivity background. In addition, ^{232}Th is mainly an alpha and beta emitter, with low penetration power, and it displays a moderate thermal neutron absorption cross section of 7.4 b sufficient to shield the container wall from most of the residual neutrons emitted by the decaying fuel bundles. The most

interesting property is the gamma and X-ray absorption properties of ThO_2 which could be comparable to those of lead, since the atomic number of ^{232}Th (90) is higher than that of lead (82). The density of 10.03 g cm^{-3} of ThO_2 (after sintering) is comparable to that of lead (11.3 g cm^{-3}), but the mass attenuation coefficients (μ/ρ) at 1 MeV are 5.293 $cm^2 g^{-1}$ and 5.105 $cm^2 g^{-1}$ for ThO_2 and Pb, respectively, as given by the code Microshield™ (30, 31). In addition, the cost of ThO_2 is expected to be quite low considering that thorium is three to five times more abundant than uranium in the Earth's crust, is very often associated with uranium in the same ores and is discarded among the mine tailings at uranium mines concentrators.

MECHANICAL STRENGTH

The mechanical stress applied to the container is evaluated on the basis of a 13 MPa constant external pressure as assessed in the AECL proposal (3), and the calculations (32) aimed at determining the minimal wall thickness that would resist to such external pressures in the case of an empty container. Of course, in an actual application, the container would be filled with not only the spent fuel bundles, but also the filler material and possibly partition walls, all of which would provide resisting forces on the container walls counteracting the effects of the external pressure. Two boron fibre/polymer mixtures (50% and 70% mass) were considered for this study and the container wall thickness was varied from 16.99 to 25.97 mm for the four polymers considered. Additional calculations indicated that, for these wall thicknesses, these materials could easily withstand the stresses from the container payloads, their strengths being, in most cases, of two orders of magnitude greater than the required strengths.

The average container inside diameter (0.645 m) and the length (2.246 m) for the containers proposed by the AECL were used, along with the 13 MPa external pressure determined in the AECL proposal. Using the respective strengths of the materials considered for the fabrication of the container, the minimum thickness needed for the basic right cylinder to withstand the pressure effects of the plutonic rock were calculated by solving the usual axial and hoop stress equations based on the thin-walled vessel assumption and an internal constant pressure pushing out on the container. In a second step, a reversal of external pressure and compressive properties was executed, using the Poisson's ratio for the materials. In a third step, the total strength of the composite (polymer + boron fibres) was determined as a simple volume-weighted average of the strength of the fibre and that of the polymer. The fourth part of the pressure calculations took in consideration the load applied to the container during the transportation of the container to the vault, as the container itself must be strong enough to withstand the weight of the 72 fuel bundles and the filler material. This weight is 1880 kg when thorium dioxide is used as filler. Table I below presents the results of these calculations.

DETERMINATION OF RADIATION DOSE ON THE CONTAINER WALL

An important part of the research investigated the resistance to radiations of the composite materials under consideration. Resistance to radiations was one of the main

reasons high polymer composites were not retained at first for the fabrication of the container, but new polymers have recently appeared on the market that are indeed much resistant to radiations. In a first part, dose calculations were carried out using the computer program Microshield™ Version 3.0 (30) in order to determine the maximum dose rate on the wall (at mid-plane) of the container from the 72 spent fuel bundles which were assumed to have been placed in the container after a 10-year cool-down period in the storage pool at the nuclear generating station following discharge from the nuclear reactor.

For these calculations, the most important radionuclides in terms of activities were included in the source term of the model. The CANDU fuel bundle investigated in this work is the standard 37-element Bruce "A" reactor bundle which has been exposed to a thermal neutron flux of 1.26×10^{14} neutrons cm^{-2} s^{-1} during 228.72 days, thus having accumulated a burnup of 685 GJ kg^{-1} initial U (7928 MW-days $tonne^{-1}$ initial U). The activities of these radionuclides were taken from an AECL publication (33) and actinides were retained for the calculations if their activities after 10-year decay was larger than 10^{-5} Ci kg^{-1}, whereas the fission products retained were those for which the activity was more than 10^{-3} Ci kg^{-1}. The Microshield™ model of the container included two different buffer materials: glass beads as proposed by AECL, and thorium dioxide (ThO_2). Typical dose rates of 11 Gy h^{-1} in the container wall were calculated with glass beads as filling material, and of about 0.05 Gy h^{-1} when thorium dioxide serves as filling material. Calculations without any filling material were also carried out and yielded dose rates at the container wall of about 19 Gy h^{-1}. Table II presents the detailed results of the Microshield™ calculations. The results indicate that the ThO_2 filler absorbs most of the gamma radiations leaving the walls of the container well shielded. Many of the results in the last column of Table II are negative, indicating that, for the materials of the container wall, the Compton effect is more important than the photoelectric and the pair production effects, giving an important built-up of photons (as calculated by Microshield™ using the conservative Taylor option). In reality, the dose absorbed within the container walls for these materials would be quite small. In this work, Microshield™ was used with the other build-up calculations options (GP and no build-up effect (B=1)), and the Taylor's method option yielded results comparable to those of the GP method, although slightly conservative. Finally, since thorium is a naturally radioactive isotope itself, an additional calculation was made to estimate its contribution to the radiation dose sustained by the container wall: this was found to be very negligible indeed.

RESISTANCE OF MATERIALS TO RADIATIONS

The research then looked at determining how the composite materials were capable to sustain such doses without failing. Samples of the four types of materials were irradiated in the pool of the SLOWPOKE-2 nuclear reactor at RMC-CMR for various durations ranging from 8 hours to more than 80 hours . Various tests were carried out on irradiated dog bone-shaped samples and the results were compared with those for unirradiated samples (22-24, 26-27). The dose rate at the irradiation site has been determined at $3.7 \pm 1. \times 10^4$ Gy h^{-1} at half reactor power (34, 35). The doses accumulated within the samples during their irradiation in the reactor pool correspond to a 50-year storage time in the

underground vault with glass bead filling material, and to 10700 years with thorium dioxide filler, if the initial source strength remains constant. When accounting for the decay of the fission products and the actinides, this corresponds to storage times of some 60 years and infinity for the glass beads and thorium dioxide fillers, respectively.

For the Devcon #10210 epoxy, the effects of irradiation indicated an increase of the tensile strength from 34±7 MPa to 44±7 MPa after 8 hours of irradiation. Previous work (22-24, 26,27) shows that the strengthening effect due to radiation promoted crosslinking becomes eventually overcome by chain scissions as exposure progresses, resulting in a significant decrease of the tensile strength past 120 hours exposure. Several samples of PEEK of two different grades were also irradiated at half reactor power for up to 83.6 hours. Even after long exposure times, little change was observed in the tensile strength other than, in some cases, a modest increase. For the 450P grade PEEK, the tensile strength was measured at 103 MPa when unirradiated and a similar value was obtained after a dose of 3.0×10^6 Gy (83.6 hours irradiation). Unirradiated 150P grade PEEK yielded a 108 MPa tensile strength while a higher value of 109 MPa was obtained after a 83.6 hour irradiation. It is important to note here that all samples were irradiated in contact with the pool water providing an oxygenated environment which emphasizes the chain scission process and decreases the polymers' resistance to radiations.

HEAT TRANSFER

The most recent phase of this research has focussed on the heat transfer considerations. These are important for two main reasons: the surrounding environment temperature should not, according to the AECL study, exceed 100 °C, and the polymer-based materials should not be subjected to temperatures exceeding their glass transition temperatures. Heat transfer data were obtained both from an experimental measurement exercise carried out on a scale-down mock-up of the container, and from the numerical and analytical solutions of the heat transfer equations at steady-state. Table III presents the temperatures at the centre (axis) of the fuelled region and at both sides of the container wall, for a worst case scenario of a container left in ambient air at 20 °C, all at mid-plane position where the temperatures are the highest on the surface of the container. These calculations were carried out for a 73 W heat output per fuel bundle, corresponding to a 1-year cool-down period after discharge from the reactor core. This provided an extreme case , as the spent fuel would spend many more years in the storage pool at the nuclear generating station. Sensitivity analyses were done to account for the uncertainties of the heat transfer parameters taken from various sources (36, 37, 38). In particular, the error bars for the heat transfer coefficient by conduction for the "polymer" were centred on the value for polystyrene, with excessive widths to account for the domain of all polymer-based materials. These results show that the use of ThO_2 as the packing material, with its conduction coefficient much better than the one of glass beads, would contribute significantly in keeping the temperatures within the container wall within acceptable values permitting the use of polymer-based composites from the heat transfer point of view.

DISCUSSION

The analysis of various polymers for use in the fabrication of a suitable container for the long-term deep underground storage of high level radioactive waste is based on the proposal by Atomic Energy of Canada Limited and used many of the data presented in the relevant publications. The activity data presented by the AECL (33) used only three significant digits, indicating that their accuracy was on the average 0.5%. The Microshield™ calculations are fairly straightforward and based on built-in data such as the mass attenuation coefficients with a 0.1% accuracy or better. However, the build-up calculations may produce results with as much as 20% variation depending on the build-up factor option selected. A conservative estimation of the accuracy of the dose calculation by Microshield™ must then retain this 20% error on these results.

The results obtained from the irradiation of the polymers in the pool of the SLOWPOKE-2 reactor are reported with accuracies of 5% or better. The reactor power is actually maintained within 0.2% by the reactor automatic control system, which is a remarkable technical achievement. However, the accurate determination of the dose received represents a challenging problem since, even if the thermal neutron flux, the fast neutron flux and the gamma flux have been determined within 10% accuracy or better at the irradiation site. The dose-to-flux conversion factor remains an elusive target as long as the neutron flux spectrum and accurate quality factors cannot be determined with sufficient accuracy. Therefore, a 28% error bar was determined in a conservative fashion (34) for the dose determination from the irradiation durations.

The mechanical strength calculations are straightforward and all the parameters are known with good accuracy, permitting to assess the error bars on these results within 1-2% at the most. As for the heat transfer calculations, overly conservative assumptions were made for the heat transfer coefficients, yet the uncertainty on the temperature results varied by only ±22% at most. It remains that the confidence in the results obtained in this work is sufficient for a good analysis of the potential of using high polymers in the fabrication of the containers for the storage of spent CANDU nuclear fuel bundles. Provided sufficient radiation shielding is provided by the filler material used within the container, both epoxy and PEEK may be used along with reinforcing boron fibre, with PEEK neatly the best of the two materials. This may be stated with assurance since both materials were irradiated in the SLOWPOKE-2 reactor pool in contact with water and subsequently investigated with a battery of tests to assess their behaviour and resistance to radiations. It was determined in previous work (22-24, 26, 27) that when high polymers are irradiated in an oxygenated environment such as water, their mechanical strength deteriorates significantly more than when irradiated in an inert environment. Therefore, our work represents a conservative approach to the investigation of these polymers as its results would apply for a worst case scenario in which the composite container wall would be in contact with underground water when irradiated. One could easily imagine protective coatings as part of the container wall fabrication which would need to keep the composite material wall dry for 50 to 100 years only, after which the radiation level would decrease to level too low to cause significant damage to the container wall, even if the material becomes

in contact with water.

An important point remains here to be discussed: that is the density of the ThO_2. In the Microshield™ model, the density of sintered thorium dioxide was used (10.03 g cm^{-3}). Assuming that sintering of thorium dioxide pellets could be compared to that of uranium dioxide pellets, it is known that the sintering operation increases the density of the compacted UO_2 pellet by about 25% so that sintering ThO_2 would have a comparable effect. In this work, the filler material for the fuelled region was taken as glass beads, although non-sintered ThO_2 as a packed powder could be used for this application. As for the packing material, sintered ThO_2 could well be used either as interlocking briquettes or donuts. A less costly approach for the ThO_2 application for the packing material could also be as a compacted powder: the density of the thorium dioxide powder would rather be of the order of 8 g cm^{-3}. The dose rates within the container would then be a bit higher than those calculated here (by a factor of approximately two), but still very much acceptable.

CONCLUSIONS AND RECOMMENDATIONS

These results represent the first phase of a more extensive research aimed at evaluating the possibilities of using high polymer composite materials for the fabrication of the high level radioactive waste storage container. There are clear indications that both PEEK and Devcon #10210 epoxy are promising candidates for this kind of application, especially when the design of the storage container provides radiation shielding materials to protect the container wall in the form of thorium dioxide. Further tests are needed on these materials to ascertain their resistance and their properties under radiation. In particular, the resistance to corrosion, which was only considered up to now from literature survey and found to be comparable if not superior to that of the metals considered in the AECL proposal, needs to be confirmed in laboratory testing which would involve ageing techniques. In particular, more knowledge is needed on water ingression within some composites. A crude cost analysis was carried out revealing that the high cost of PEEK (about $110/kg) and that of the boron fibres ($275/kg) may represent a significant drawback, but it is expected that increased production to meet a larger demand for this product may well drive the prices down within a few years, making the economics indeed favourable to the point of making the cost of producing a polymer (PEEK)-based container with thorium dioxide filler comparable to the cost of a similar container based on titanium alloy for the shell and using glass beads for the filling material.

REFERENCES

1. "Summary of the Environmental Impact Statement on the Concept for Disposal of Canada's Nuclear Fuel Waste", AECL-10721, 1994.

2. "The Performance, Assessment and Ranking of Container Design Options for the Canadian Nuclear Fuel Waste Management Program", AECL-TR-500, 1994.

3. "Environmental Impact Statement on the Concept for Disposal of Canada's Nuclear Fuel Waste", AECL-10711, 1994.

4. "The Disposal of Canada's Nuclear Fuel Waste: Engineered Barrier Alternatives", AECL-10718, 1994.

5. "An Assessment of the Feasibility of Indefinite Containment of Canadian Nuclear Fuel Wastes", AECL-10972, 1995.

6. "The Disposal of Canada's Nuclear Fuel Waste: A Study of Postclosure Safety of In-Room Emplacement of Used CANDU Fuel in Copper Containers in Permeable Plutonic Rock; Volume 5: Radiological Assessment", AECL-11494-5, 1996.

7. Dorgham M.A., "Designing with Plastics and Advanced Plastic Composites", Warwick Printing Co., U.K.,1986.

8. Feldman D., "Polymeric Building Materials", Elsevier, New York, 1989.

9. "Designing with Composite Materials", Institute of Mechanical Engineers, J.W. Arrowsmith Tld., U.K., 1973.

10. Kawata K. and Akasaka T., "Composite Materials: Mechanics, Mechanical Properties and Fabrication", Applied Science Publishers Ltd., Essex, U.K., 1982.

11. Kroschwitz J.I., "Concise Encyclopedia of Polymer Science and Engineering", Wiley, New York, 1990.

12. Margolis J.M., "Advanced Thermoset Composites: Industrial and Commercial Applications", Van Nostrand Reinhold, New York, 1986.

13. Mark H.F., Bikales N.M., Overberger C.G., and Menges G., "Encyclopedia of Polymer Science and Engineering, Wiley, New York, 1985.

14. McCram N.G., Buckley C.P., and Bucknall C.B., "Principles of Polymer Engineering", Oxford University Press, New York, 1988.

15. Piatti G., "Advances in Composite Materials", Applied Science Publishers, Essex, U.K., 1978.

16. Rodriguez F., "Principles of Polymer Systems", 3rd Ed., Hemisphere Publishing Corp., New York, 1989.

17. Schwartz M.M., "Composite Materials Handbook", McGraw-Hill, St.Louis, 1984.

18. Schweitzer P.A., "What Every Engineer Should Know about Corrosion", Vol 21, Marcel Dekker, New York, 1987.

19. Weeton J.W. & Scala E., "Composites: State of the Art", The Metallurgical Society of AIME, New York, 1974.

20. Callister W.D., "Materials Science and Engineering: An Introduction", 2nd Ed., Wiley, New York, 1991.

21. Minns D., McIntyre N.S., Fadihy T.Z., Landsberger S. & Blenkiron A.L., "The Chemical Institute of Canada: Assessment of AECL's Environmental Impact Statement on the Concept for Disposal of Canada's Nuclear Fuel Waste", The Chemical Institute of Canada, Ottawa, Ont., 1995.

22. Poirier P.E., "Effets des Radiations sur les Époxies pour Composites Polymériques", Document RMC-CCE-CM417-95-6, Royal Military College of Canada, 1995.

23. Bonin H.W., Bui V.T., and Poirier E., "Effects of Neutrons and Gamma Radiation of High Polymer Epoxy Adhesives", Proc.16th Ann. Conf. of Can. Nucl. Soc., Vol I, Saskatoon, Sask., 4-7 June 1995.

24. Harris H.C., "Radiation Effects on Polymer Composite Epoxies", Document RMC-CCE-CM417-96-16, Royal Military College of Canada, Kingston, Ont., 1996.

25. Victrex™ PEEK Materials Properties.

26. Pagé J.Y.S.D., "Effects of Neutron and Gamma Radiation on the Viscous Behaviour of Semi-Crystalline PEEK", Master's degree thesis, Dept. Chemistry & Chemical Engineering, Royal Military College of Canada, 1997.

27. Pagé J.Y.S.D., Bonin H.W., Bui V.T. & Bates P., "Neutron and Gamma Radiation Effects on the Viscosity Behaviour of Poly (Aryl Ether Ether Ketone)", Proc. 18th Ann. Conf. Can. Nucl. Soc., Toronto, Ont., 8-11 June 1997.

28. Watt W. and Perov B.V., "Strong Fibres", Vol 1, Elsevier Science Publishers, Amsterdam, Netherlands, 1985.

29. Marmer E.N., "High Temperature Materials", Freund Publishing House, Israel, 1971.

30. "Microshield: Version 3 User's Manual", Grove Engineering Inc., Rockville, Md, USA, 1988.

31. "Microshield: Version 5 User's Manual", Grove Engineering Inc., Rockville, Md, USA, 1996.

32. Budynas R.G., "Advanced Strength and Applied Stress Analysis", McGraw-Hill, New York, 1977.

33. "Derivation of Initial Radionuclide Inventories for the Safety Assessment of the Disposal of Used CANDU Fuel", AECL-9881, 1989.

34. Lamarre G.B., "Experimental and Computational Determination of Radiation Dose Rates in the SLOWPOKE-2 Research Reactor at the Royal Military College of Canada / Collège Militaire Royal du Canada"; M. Eng. Thesis, Royal Military College of Canada / Collège Militaire Royal du Canada, Kingston, Ontario, April 1999.

35. Lamarre G.B. and Bonin H.W., "Experimental and Computational Determination of Radiation Dose Rates in the SLOWPOKE-2 Research Reactor at the Royal Military College of Canada"; Proc. 20th Ann. Canadian Nuclear Society's Conf., Montréal, Qué., Canada, 30 May-2 June 1999, Session B3.

36. Caruso *et al.*, "Matériaux Nucléaires - Tome I", École Polytechnique de Montréal, Montréal, Qué., Canada, 1970.

37. Glasstone S. and Sesonske A., "Nuclear Reactor Engineering", 4th Ed., Chapman & Hall, New York, 1994.

38. Kreith F. and Bohn M.S., "Principles of Heat Transfer", 5th Ed., West Publishing Co., St. Paul, MN, USA, 1993.

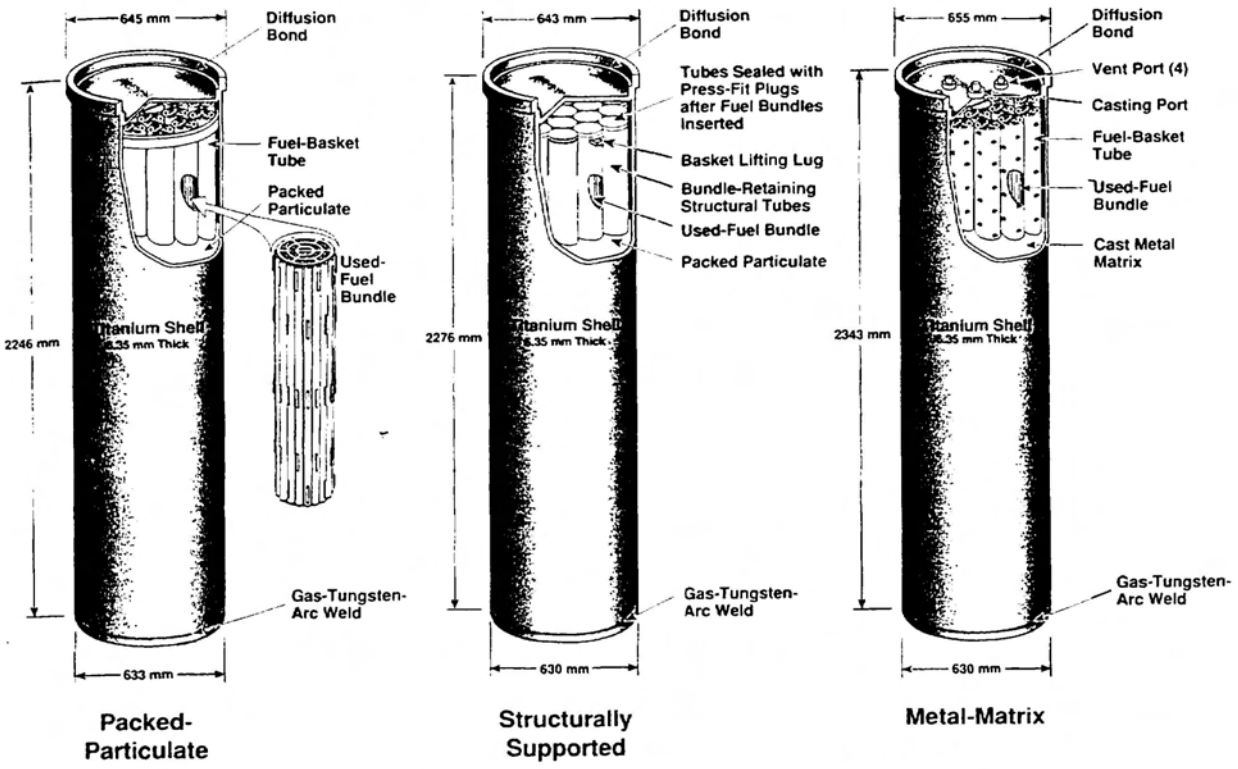

Figure 1: Spent Fuel Storage Container (3).

TABLE I: STRENGTH OF COMPOSITE MATERIALS AND THICKNESS OF CONTAINER WALL

COMPOSITE MATERIAL (% by mass)	TENSILE STRENGTH (MPa)*	REQUIRED TENSILE STRENGTH (MPa)	COMPRESSIVE STRENGTH (MPa)	WALL THICKNESS OF CONTAINER (mm)
PS with 50% boron	41.9	0.337	161	26
PS with 70% boron	49.1	0.425	207	20
PMMA with 50% boron	51.5	0.372	178	24
PMMA with 70% boron	57.0	0.477	226	19
EPOXY with 50% boron	52.0	0.432	205	20
EPOXY with 70% boron	57.6	0.523	247	17
PEEK with 50% boron	88.1	0.406	193	22
PEEK with 70% boron	82.6	0.506	239	18

* : The tensile test data for PS and PMMA are based on the minimum tensile strengths measured using the 28.5 hour irradiation data. Those for the Devcon 10210 epoxy and the PEEK are based on the 8- and 80-hour irradiation in the SLOWPOKE-2 reactor pool.

TABLE II : Microshield™-CALCULATED DOSES IN CONTAINER WALLS

CONTAINER MATERIAL	FILLER	DOSE RATE INSIDE WALL Gy h^{-1}	DOSE RATE OUTSIDE WALL Gy h^{-1}	ABSORBED DOSE IN SHELL Gy h^{-1}
TITANIUM	NONE	14.5	15.2	-0.66
TITANIUM	GLASS BEADS	11.2	7.14	4.02
TITANIUM	ThO_2	0.0517	0.0573	-0.00565
COPPER	NONE	14.5	4.98	9.50
COPPER	GLASS BEADS	11.2	1.83	9.34
COPPER	ThO_2	0.0517	0.014	0.0379
PS-50% BORON	NONE	14.5	18.6	-4.65
PS-50% BORON	GLASS BEADS	11.2	10.9	-0.180
PS-50% BORON	ThO_2	0.0517	0.0917	-0.0400
PS-70% BORON	NONE	14.5	19.0	-4.52
PS-70% BORON	GLASS BEADS	11.2	11.2	-0.06
PS-70% BORON	ThO_2	0.0517	0.0944	-0.427
PMMA-50% BORON	NONE	14.5	18.6	-4.14
PMMA-50% BORON	GLASS BEADS	11.2	10.4	0.72
PMMA-50% BORON	ThO_2	0.0517	0.0872	-0.0355
PMMA-70% BORON	NONE	14.5	19.1	-4.6
PMMA-70% BORON	GLASS BEADS	11.2	11.1	0.72
PMMA-70% BORON	ThO_2	0.0517	0.0931	-0.0414
EPOXY-50% BORON	NONE	14.5	19.1	-4.58
EPOXY-50% BORON	GLASS BEADS	11.2	10.5	0.67
EPOXY-50% BORON	ThO_2	0.0517	0.0871	-0.0354
EPOXY-70% BORON	NONE	14.5	19.3	-4.77
EPOXY-70% BORON	GLASS BEADS	11.2	11.0	0.13
EPOXY-70% BORON	ThO_2	0.0517	0.0924	-0.0407
PEEK-50% BORON	NONE	14.5	19.0	-4.46
PEEK-50% BORON	GLASS BEADS	11.2	11.0	0.16
PEEK-50% BORON	ThO_2	0.0517	0.0923	-0.0406
PEEK-70% BORON	NONE	14.5	19.1	-4.65
PEEK-70% BORON	GLASS BEADS	11.2	11.3	-0.18
PEEK-70% BORON	ThO_2	0.0517	0.0957	-0.0440

Table III : HEAT TRANSFER DATA (Analytical Solution)

FUELLED REGION	PACKING MATERIAL	CONTAINER WALL MATERIAL	TEMPERATURE AT THE CENTRE OF THE FUELLED REGION	TEMPERATURE AT THE INSIDE SURFACE OF THE CONTAINER WALL	TEMPERATURE AT THE OUTSIDE SURFACE OF THE CONTAINER WALL
UO_2 + GLASS BEADS	GLASS BEADS	TITANIUM	585 ± 47 K	345 ± 2 K	345 ± 2 K
UO_2 + GLASS BEADS	GLASS BEADS	COPPER	585 ± 47 K	345 ± 2 K	345 ± 2K
UO_2 + GLASS BEADS	GLASS BEADS	POLYMER	651 ± 130 K	410 ± 90 K	365 ± 4 K
UO_2 + GLASS BEADS	ThO_2	TITANIUM	412 ± 9 K	345 ± 2 K	345 ± 2 K
UO_2 + GLASS BEADS	ThO_2	COPPER	411 ± 10 K	345 ± 2 K	345 ± 2 K
UO_2 + GLASS BEADS	ThO_2	POLYMER	480 ± 30 K	410 ± 90 K	363 ± 4 K

SIMULATION PARAMETERS

Volumetric heat production Q = $1.39 \pm .01 \times 10^4$ W m^{-3} (Corresponds to 73 kW per bundle, after 1 year storage after discharge from the reactor core).

Fuelled region radius a = 0.23175 m
Outer radius of packing material region b = 0.31015 m
Outer radius of container wall c = 0.3165 m

Height of inside cavity of container H = 2.246 m

Heat transfer coefficient by conduction for fuelled region k_1 = 3.49 ± 0.35 W m^{-1} K^{-1}
Heat transfer coefficient by conduction for packing region (glass beads) k_2 = 0.58 ± 0.1 W m^{-1} K^{-1}
Heat transfer coefficient by conduction for packing region (ThO_2) k_2 = 8.37 ± 1.0 W m^{-1} K^{-1}
Heat transfer coefficient by conduction for container wall (Titanium) k_3 = 22.0 ± 1.0 W m^{-1} K^{-1}
Heat transfer coefficient by conduction for container wall (Copper) k_3 = 399.0 ± 10.0 W m^{-1} K^{-1}
Heat transfer coefficient by conduction for container wall (Polymer) k_3 = 0.16 ± 0.1 W m^{-1} K^{-1}

Heat transfer coefficient by convection for container wall (Titanium and Copper) h = 22.7 ± 1.0 W m^{-2} K^{-1}
Heat transfer coefficient by convection for container wall (Polymer) h = 16.9 ± 1.0 W m^{-2} K^{-1}

Temperature of ambient air T_{air} = 293.16 K