IAEA-TECDOC-375

INTERNATIONAL STUDIES ON CERTAIN ASPECTS OF THE SAFE TRANSPORT OF RADIOACTIVE MATERIALS, 1980 – 1985

REPORT OF THE CO-ORDINATED RESEARCH PROGRAMME
ON SAFE TRANSPORT OF RADIOACTIVE MATERIALS
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FOREWORD

The International Atomic Energy Agency has been involved in promoting the safe transport of radioactive materials since the late 1950's. The Agency first published the **Regulations for the Safe Transport of Radioactive Materials, Safety Series No. 6**, in 1961, and these regulations have been periodically reviewed and updated since that time. In order to guide the reviews of the regulations, data on certain aspects of transport were determined to be needed. To assist in encouraging the development and exchange of such data, the Agency approved the Coordinated Research Programme on Safe Transport of Radioactive Materials in September 1979, and the first research agreement pertaining to the programme was finalized in February 1980.

Through the end of the programme, it involved three research contracts and eight research agreements. The present document contains final summary reports on the activities performed during the entire coordinated research programme, from its inception in 1980, through two intermediate Research Coordination Meetings in Albuquerque, USA, June 1981, and in Rome, Italy, March 1983, to its concluding Research Coordination Meeting in London, UK, July 1985 (RC-206.3). Although RC-206.3 was the last meeting for the Coordinated Research Programme on Safe Transport of Radioactive Materials, some of the individual agreements will continue to their normal expiry dates and the remaining contract (with the Philippines) may be extended beyond its current expiry date. Work in many of these areas will continue beyond the expiration of the research agreements.

In summary, through the CRP, (1) the basis for activity limits used in the 1985 Edition of Safety Series No. 6 was developed, (2) the understanding of the structural and thermal response of transport packages to various environments was extended, (3) a generic risk assessment model (INTERTRAN) was developed, and (4) the risks and exposures resulting from radioactive material transport were investigated.

EDITORIAL NOTE

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INTRODUCTION

In order to encourage research into certain aspects of the safe transport of radioactive materials specifically relating to the regulations and regulatory standards, and to assist in the international exchange of such information, the International Atomic Energy Agency initiated the Coordinated Research Programme on Safe Transport of Radioactive Materials in early 1980. The data resulting from this programme assisted in the technical review of the 1973 (As Amended) Regulations for the Safe Transport of Radioactive Materials, Safety Series No. 6, which led to the 1985 Edition of Safety Series No. 6. The data will also assist in future revisions of Safety Series No. 6 and will be useful in guiding proper implementation of the Regulations.

At the beginning of the programme, the following programme goals were envisaged:

(a) develop methods of assessment of individual and collective doses to transport workers and members of the public during the transport of radioactive materials, particularly those related to the transport of irradiated fuel casks;

(b) develop methods of assessing leakage rates from large transport packages; and

(c) determine activity limits for radioactive material transport packagings.

The first research agreement was finalized in February 1980, and through the duration of the Coordinated Research Programme, a total of eight research agreements and three research contracts were consummated.

The goals outlined above were met as follows. The Swedish research agreement has resulted in the development of a basic model code, INTERTRAN, for assessing radiological impact associated with the transport of radioactive materials. A user's manual has likewise been prepared, and was published by the Agency as TECDOC-287. INTERTRAN is being used by Member States to make individual assessments of the radiological impact of transporting radioactive materials. Those Member States which currently have the INTERTRAN code include Austria, Australia, Brazil, Canada, Egypt, Finland, France, FRG, Greece, Italy, Japan, Libya, the Netherlands, the Philippines, Sweden, Turkey, UK and USA. The development and application of INTERTRAN was also supported by the USA Research Agreement. These activities addressed programme goal a.

The studies in Argentina, India, Italy and the Philippines, and part of the studies in France, the UK and the USA have all related to the assessment of individual and collective doses to transport workers and members of the public during the transport of radioactive materials, also addressing programme goal a.

The Canadian and GDR studies and part of the French study have provided valuable data on the response of packagings and their contents to various normal and possible accident environments, and to this end these activities will assist in calibrating the analytical models used in INTERTRAN. Thus, these activities have indirectly supported the programme goal a, and to some extent, have supported programme goal b. Specifically, the French study on long term fires has provided information on the performance of package containment and shielding systems in these very adverse conditions. Likewise, the Canadian study has provided data on the behaviour of irradiated nuclear fuel in shock and vibration environments. The research in the GDR has been looking into the heat removal and heat transfer in a model transport cask, which will provide a better understanding of the mechanism of heat transfer in a transport cask, permitting a more accurate prediction of fuel temperature and the possibility of radioactive material release.

Under the UK research agreement a "Q-system" for calculating the content limits for Type A packages has been developed and refined. This was used as a basis to determine A_1 and A_2 activity limit values for inclusion in the 1985 Edition of the Transport Regulations, Safety Series No. 6. This addressed programme goal c.

Although the Coordinated Research Programme on Safe Transport of Radioactive Materials has been formally concluded, some of the individual agreements and contracts will continue to their normal expiry dates and work in many of these areas will continue beyond the expiration of the research agreements.

Through the CRP, (1) the basis for activity limits used in the 1985 Edition of Safety Series No. 6 was developed, (2) the understanding of the structural and thermal response of transport packages to various environments was extended, (3) a generic risk assessment model (INTERTRAN) was developed, and (4) the risks and exposures resulting from radioactive material transport were investigated. Section II of this report provides a summary of each research agreement or research contract which was part of the Coordinated Research Programme. Section III lists reports resulting from the programme. Finally, Section IV provides detailed summary reports for each activity.

SUMMARY OF CONTRACTS AND AGREEMENTS

International Atomic Energy Agency Project Officers:
1980-1983 B.C. Bernardo
1983-1985 R.B. Pope

Research Agreement No. 2291/R3/CF

Member State: UK

Institute: Central Electricity Generating Board, Berkeley Nuclear Laboratories, Berkeley, Gloucestershire, UK

Title of Project: **Activity Release Limits for Irradiated Fuel Transport Flasks**

Chief Scientific Investigator: B.M. Wheatley

Other Scientific Staff: H.F. Macdonald, E.P. Goldfinch

Expiry: 1985

Abstract:

The initial phase of work under this Research Agreement was devoted to an assessment of the individual and collective doses associated with the routine transport of irradiated fuel from commercial nuclear power stations in the UK. The levels of radiation dose involved to both transport workers and members of the public were shown to be low in comparison with statutory limits and with other sources of dose arising in the nuclear fuel cycle. In addition, the dosimetric implications of the Type B package activity release limits contained in the 1973 IAEA Transport Regulations were examined. It was shown that the Type B(U) limit under accident conditions appeared unduly restrictive in comparison with safety standards commonly applied at power reactor sites and in the 1985 revised Regulations this limit is relaxed such that the earlier Type B(M) limit is now applied to all Type B packages. Also, methods of evaluating the radioactive source term in transport accident assessments were considered and attention drawn to the potential significance of the temporal variation in the source strength and the respirable aerosol fraction of the released material as factors which could minimize the radiological impact of an accident.

The later phases of this work were concerned with the development of the Q system which is a refinement and extension of the earlier A_1/A_2 system for the determination of Type A package contents and other limits within the IAEA Transport Regulations. Following review by a Special Working Group meeting convened by the Agency, the Q system was adopted for inclusion in the 1985 revised Regulations and will be fully described in the revised version of Safety Series No. 7. It incorporates the latest recommendations of the ICRP and by explicitly identifying the dosimetric considerations underlying the derivation of the A_1 and A_2 values provides a firmer and more defensible basis for the Regulations. Overall, the adoption of the Q system has introduced only relatively minor variations in the A_1 and A_2 values compared with those of the 1973 Regulations. This is also true of activity release limits for Type B packages which were examined in relation to the implications of the revised Regulations on radiological limits in the transport of irradiated nuclear fuels.

Research Agreement No. 2620/R1/CF

Member State: Sweden

Institute: KEMAKTA Konsult AB, Stockholm, Sweden

Title of Project: **World-wide Risk Assessment of the Transportation of Radioactive Materials**

Chief Scientific Investigator: A-M. Ericsson

Other Scientific Staff: P. Ek, B. Dufva, M. Elert

Expiry: 1982

Abstract:

The following* provides an up-to-date summary, as of July 1985, of the status of the computer code INTERTRAN: A System for Assessing the Impact from Transporting Radioactive Material. The computer code was developed as part of a cooperative Research Agreement with the Swedish Nuclear Power Inspectorate, initiated in 1980.

Section IV.C of this document contains a summary report prepared by the principle individuals involved in the development of the code, which was published in the proceedings of the 7th International Symposium on Packaging and Transportation of Radioactive Materials (PATRAM '83), May 15-20, 1983, New Orleans, Louisiana, USA, CONF-830528--Vol. 1. This summary report outlines the capabilities of the code.

The development of the code was guided by an Oversight Committee which included members from eight countries. Following the development of the code, a detailed report and users manual was prepared and subsequently published by the International Atomic Energy Agency (IAEA) in 1983.**

During 1983, the users manual** was made available to all Member States of the IAEA, and it was suggested that these governments consider its usefulness for carrying out assessments of the radiological impact of transporting radioactive material. Based upon the concerns of experts however, it was not suggested that it be used for a global assessment as indicated by the title of the Research Agreement, since it was indicated that results therefrom should be used with caution "because of possible inaccuracies in the input data and intrinsic uncertainties in the methodology".

During the time the code has been under practical scrutiny and in practical use the need for some corrections to the code has been identified. A meeting of interested parties was held in Stockholm, Sweden in June 1984 aiming at specifying the areas where corrections, rather than refinements, of the code were necessary and appropriate. These corrections were introduced into the code in late 1984.

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* This text has been prepared by Mr. R.B. Pope, IAEA and Mr. B.G. Pettersson, Swedish Nuclear Power Inspectorate.

** Ericsson, A-M, and Elert, M., "INTERTRAN: A System for Assessing the Impact from Transporting Radioactive Material", IAEA-TECDOC-287, International Atomic Energy Agency, Vienna, Austria, 1983.

The code has been placed in the OECD Nuclear Energy Agency Data Bank in France and it is also available from the computer code libraries located at the Argonne National Laboratory and the Oak Ridge National Laboratory in the USA. It is available in two versions -- IBM and CDC. The code has been made available to institutions in 19 countries.

Research Agreement No. 2715/R1/CF

Member State: Canada

Institute: Ontario Hydro, Toronto, Canada

Title of Project: **Investigation into the Ability of Irradiated CANDU fuel to Withstand Transportation Shock and Vibration Environment**

Chief Scientific Investigator: D.W. Souther, R.C. Oberth

Other Scientific Staff: J.F. Tanaka, D.J. Knight, T. Loewen, J.W. Forest

Expiry: 1984

Abstract:

This activity, which spanned a period from 1975 to 1984, was undertaken with the primary objective of evaluating the response of irradiated CANDU fuel to the shock and vibration transportation environment. Concurrently, data from field tests of road and rail transport systems similar to those which might be used for transporting irradiated fuel casks were obtained. These data provide guidance on transient and steady-state acceleration levels and power spectral densities which can be used to guide design and testing efforts relating to large cask systems.

For the field tests, a 32 tonne mass containing a simulated spent fuel module was placed on a tractor/trailer system having a combined, unloaded mass of 18 tonne; and a similar 68 tonne mass was placed on a flat rail car of 28.9 tonne. Vertical components of acceleration were measured at the baseplate on which the mass was mounted and on the central module tube. For steady-state transport, maximum measured accelerations at the baseplate were 0.18 g for road and 0.20 g for rail; whereas, on the module tube the maximum values were 0.45 g for road and 0.30 g for rail. Power spectral densities showed most steady-state vibration energy is in the 0 to 300 Hz range for road and in the 0 to 500 Hz range for rail. Transient environments, such as a truck crossing railway tracks and, for railcars, the take-up of coupler slack or the crossing of switches produces amplifications in these values by factors of 1.1 to 3.4 for road transport and by as much as a factor of 10 for rail transport.

The CANDU fuel bundles were found to be highly resistant to impact shocks. No fuel bundle failures were found during testing for axial impact loads of 175 g and lateral impact loads of 78 g for axial-restrained bundles. CANDU fuel bundles were determined to be adequately resistant to fatigue cracking, mitigating concerns of fuel bundle disassembly. However, for very poor roads, vibration levels that could exceed the threshold were predicted. Thus extended travel on very poor roads is to be avoided.

This study provided data for guiding system design. For example, the CANDU fuel bundles need not be axially restrained within the cask cavity during road or rail transport. The two module stack within the cask cavity need not be restrained vertically. Bundle failures preventing automated fuel

handling are not expected. Rail cars carrying a fuel cask should have a coupler with a stroke of at least 255 mm. Hydraulic type draft gears produce the lower acceleration levels over friction types. Finally, the effective stiffness of the tiedown system, over a 10^7 to 10^9 N/m range, has little effect on the response of the system during rail car coupling impact.

Research Contract No. 2740/R1/RB

Member State: India

Institute: Bhabha Atomic Research Centre, Trombay, Bombay, India

Title of Project: **Methods of Assessment of Individual and Collective Doses to Transport Workers and Members of the Public During the Transport of Radioactive Materials**

Chief Scientific Investigator: K.G. Vohra

Other Scientific Staff: G. Subrahmanian, A.N. Nandakumar, R.K. Kher, S.R.K. Iyer

Expiry: 1983

Abstract:

Transport workers handling radioactive cargoes are generally exposed to the highest dose rates of any population group. Methods of assessment of dose received by transport workers are studied to arrive at a useful method. An empirical model based on a detailed work study of individuals handling radioactive cargoes and the exposure rates at various distances from specific individual packages is developed. The personnel doses thus calculated compared reasonably well with the doses recorded on personnel monitoring badges. The personnel doses were also evaluated with reference to the total transport index handled by the workers, yielding results consistent with those reported elsewhere by earlier researchers.

For assessing the collective dose to the public due to urban transport of radioactive material, the space around the vehicle transporting cargo was divided into a number of cells of dimensions 1 m x 1 m. The radiaition level in each cell was measured and the pedestrian density along the route was obtained. Using the pedestrian occupancy in the cells and the measured radiation levels, the total dose to the public was assessed. A similar assessment was made with respect to the passengers in the neighbouring vehicles. The suggested method of calculation may aid determination of the route and time of transport and the preferable traffic configuration for the vehicle carrying the radioactive consignments for optimizing the dose to the urban public.

Research Agreement No. 2792/R2/CF

Member State: France

Institute: C.E.A., Paris, France

Title of Project: **Evaluation of Safe Transport of Radioactive Materials with Respect to Fires of Long Duration and with Respect to Integrated Doses Received by Personnel**

Chief Scientific Investigator: Y. Sousselier and C. Ringot

Other Scientific Staff: J.C. Nenot

Expiry: 1985

Abstract:

This study consisted of two components. The first component was related to the evaluation of safety margins of packagings in severe accidental conditions involving fire environment. Different types of packagings are being studied as follows:

1. <u>Spent-fuel packaging for PWR irradiated fuel elements</u>

Many transports of PWR fuel elements to the French reprocessing plant of La Hague are done by sea. It is important to have a good knowledge of the behaviour of these packagings in case of a non-controlled fire on board a ship.

2. <u>PuO_2 packaging</u>

For the transport of PuO_2 powder, a special packaging Type B(U)F was recently developed in France. Taking into account the possibility to transport PuO_2 by air, the behaviour of these packages under very severe fire conditions (1000°C) has been evaluated.

3. <u>UF_6 packagings</u>

In case of a non-controlled fire involving a cargo of UF_6 packagings on board a ship, the consequences may be very severe due to the high toxicity of this material. Ship fires cover a large scale of temperature and duration conditions. With conservative assumption, calcuations were performed some years ago which have demonstrated a certain level of safety for the 48Y packagings which are commonly used for the transportation of natural UF_6 - the time before rupture of the vessel was evaluated at 60 minutes for an 800°C fire and at 45 minutes for a 900°C fire. Following the Mont Louis accident, it was decided to reevaluate this margin of safety with more realistic assumptions and with a large scale of conditions (200°C to 800°C, and 30 minutes to 48 hours).

4. <u>Transport of small sources (Mo-99)</u>

<u>"Type A" packaging "ELUMATIC III"</u>

The transportation of Type A, not as special form radioactive material, is subject to strict rules in the tunnel traffic of some international tunnels, like the tunnel of Mont Blanc and tunnel de Fréjus between France and Italy. The total amount of activity by vehicle must be less than A2/3. An important traffic of radioisotopes is using these tunnels, and in particular of "ELUMATIC III" packagings (technitium generators). With the value of A_2 admitted to Mo-99 in the new regulations (1985 Edition) which is only 10

curies instead of 100 curies, an economical impact is resulting. A risk assessment is being performed to reevaluate the limitation of A_2. In the framework of this study, fire tests are performed to determine the radioactive release in the environment in case of failure.

The second component of the study was related to the radioactive exposure to transport personnel involved in the handling and shipment of various types of radioactive material over a four year period. For workers involved with the packing, transport and delivery of approximately 160,000 packages of radiopharmaceutical products, radioactive sources and products for medical analysis, an average annual collective dose of 0.26 man Sv was measured, yielding an average individual dose of 15.5 mSv. The transport of PWR and BWR spent fuel packages resulted in essentially no exposure to workers. The transport of nuclear materials, such as natural or enriched uranium and plutonium in either metal or oxide forms, resulted in an average individual dose of approximately 0.25 mSv, with a maximum of 1.5 mSv, over a three year period.

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Research Agreement No. 2793/R1/CF

Member State: USA

Institute: Sandia National Laboratories, Albuquerque, New Mexico, USA

Title of Project: **Development of a Generic Risk Assessment Code for Radioactive Material Transportation**

Chief Scientific Investigator: R.E. Luna

Other Scientific Staff: B. Biringer, S. Daniel, M. Madsen, J.D. McClure, K.S. Neuhauser, R.M. Ostmeyer, P.C. Reardon, J.M. Taylor

Expiry: 1985

Abstract:

The Coordinated Research Programme agreement with the US (at Sandia National Laboratories) covered support of the development of INTERTRAN, development of risk assessment techniques and support of Member States' efforts to use INTERTRAN. While there were no requests made to Sandia for member support under the last item, significant activities were undertaken and completed to support the initial two items. Since INTERTRAN is an outgrowth of RADTRAN, US activities to expand and improve RADTRAN are likely applicable to INTERTRAN. Such activities included: correction of on-leak doses, improving stop models for truck and rail, including ingestion doses, modelling multiple package movements and compiling a probability-consequence spectrum to aid in accident risk interpretation. In addition, the concept of unit risk factor (URF) as a simplifying concept which is applicable to simple risk estimation problems or multiple evaluation under different conditions has been pursued. In the URF technique the approximate risk is determined from total miles and total shipments through a URF calculated for the problem or otherwise available that converts shipment-kilometres to risk units. The technique can be applied both to normal and accident situations. In concept the URF is not unlike published accident rates, e.g., fatalities/10^5 vehicle miles which are based on occurrence of death on highways or railroads. The risk methods developed have been used for a variety of purposes in the US and several are highlighted specifically supporting the environmental assessments for The Nuclear Waste Policy Act. These analyses include Cost Risk I and II and a

paper which shows the sensitivity of spent fuel shipment risk assessment to variations in input parameters. It is expected that the US efforts will be reflected in both the future use and useability of INTERTRAN and further the IAEA's policy of increasing transport risk assessment activities of its members.

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Research Agreement No. 2837/R2/CF

Member State: Italy

Institute: ENEA, Rome, Italy

Title of Project: **Assessment of Individual and Collective Radiation Doses to Transport Workers and Population from the Transport of Radioactive Materials**

Chief Scientific Investigator: S. Piermattei

Other Scientific Staff: S. Mancioppi, G. Scarpa

Expiry: 1985

Abstract:

The paper deals with the application of the INTERTRAN code to the transport of radioactive materials on Italian territory. Particular attention is devoted to the problems connected with the transport of radioactive wastes, temporarily stored at the production sites, to one hypothetical disposal site. In fact, the exposure of the individuals involved is an important parameter to be considered in order to choose among different hypothetical sites located on the territory.

To perform such an evaluation the INTERTRAN code was used.

Two different modes of transport: truck and rail plus truck, have been considered.

The application of the code is limited to normal conditions of transport.

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Research Agreement No. 2954/R1/CF

Member State: German Democratic Republic

Institute: VE Kombinat, Kernkraftwerke, "Bruno Leuschner", Greifswald, German Democratic Republic

Title of Project: **Heat Transfer Investigations within Dry Spent Fuel Casks**

Chief Scientific Investigator: F. Nitsche

Expiry: 1986

Abstract:

For the investigation of the heat transfer within dry spent fuel casks model experiments were performed in which the spent fuel assemblies (SFA) and spent fuel elements (SFE) were simulated by electrically heated rods. The maximum heater surface temperatures were measured in dependence on various

parameters and compared with calculations. The model experiments were performed in 2 stages:

(1) Investigation of the heat transfer from 30 SFAs to the inner cask wall using a model arrangement of 30 heating rods in a model cask at a scale of 1:8.

(2) Investigation of the heat transfer within a SFA using a SFA model of a pressurized-water type reactor with 90 heating rods.

The results obtained in the first stage are represented in the Report SAAS-306 (Berlin, 1983). They allowed the maximum SFA surface temperature to be determined and led to a better qualitative in-sight in the heat transfer processes in vertical and horizontal cask position, to a more precise evaluation of internal temperatures measurable in the original cask and to a quantitative description of the radiant heat transfer between the SFAs and the cask wall. In the present second stage the temperature distribution of the SFA model is measured for different heat rates, in free air environment, in the model cask under vacuum and also under normal pressure and overpressure with different cooling gases (air, argon, helium). In free air environment roughly the same maximum heater surface temperatures were measured in horizontal and vertical SFA positions.

Based on the measurements under vacuum a computer programme is validated and a simplified analytical representation is derived for calculating the radiant heat transfer in the SFA. The results obtained so far for the investigations in the air-filled, pressurized (0.1 ... 0.7 MPa) model cask show that the heat removal is considerably increased by pressure increase above all in the pressure range from 0.1 to 0.5 MPa and that the convective heat transfer becomes the dominant process. Currently a computer programme is developed and the calculation results are compared with the measurements. The investigations will be supplemented for the coolants argon and helium and completed according to working plan of Research Agreement by 31 March 1986.

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Research Agreement No. 3073/R1/CF

Member State: Argentina

Institute: Comision Nacional de Energia Atomica, Buenos Aires, Argentina

Title of Project: **Evaluating Risk in Transport of Tritiated Water**

Chief Scientific Investigator: E. Palacios

Other Scientific Staff: A. Biaggio, C.A. Menossi, R. Segado, R. Reyes

Expiry: 1985

Abstract:

The individual risk of a programme involving the transport of tritiated water will depend on the probability for an individual to be involved in the transport accident and on the conditional probability for an individual involved in the accident to receive a given dose.

The probability for an individual to be involved in the accident is a function, among others, of the accident rate in the area under consideration and of the transport periodicity. In turn, the conditional probability for an individual involved in the accident to receive a given dose will depend on the

exposure model being considered, on the probability of occurrence of a spill with certain characteristics and on the probability of occurrence of given meteorological conditions.

The model used for evaluating the doses that would be incurred by the most exposed individuals in the case of an accident is described. The analyzed accidents were classified by their severity in Categories I through VII and only those with severity above II will produce doses in the critical group. The doses resulting from accidents with severity between categories III and VIII, whose relative frequencies range between 7×10^{-2} and 2×10^{-5}, are only modified in a factor of 2. Besides, the most unfavourable meteorological condition during normal transport operations in Argentina (high evaporation rate) would show an annual frequency above 50%. Therefore, and for practical reasons, the dosimetric factor applied for the model used was 2 mSv $Ci^{-1}\ell$, corresponding with the most unfavourable situation (accident severity VIII and high evaporation rate).

The Argentine authority has established a probabilistic criterion in the safety analysis of nuclear power plants (NPPs) based on the maximum individual risk of the members of the public. A criterion was adopted for potential accidental situations, by which the individual risk of the most exposed individuals is in the same order of magnitude as that accepted for normal operations.

The acceptance criterion adopted for NPPs may be extended to the transport of radioactive materials if a probability of death of 10^{-6} per year is accepted for an entire programme of transport of tritiated water. Thus, an acceptance curve may be drawn, relating the probability of occurrence of accidents in transport with the concentration of the transported tritiated water. Finally, a hypothetical example of application is developed.

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Research Contract No. 3744/R1/RB

Member State: Philippines

Institute: Philippine Atomic Energy Commission, Diliman, Quezon City, Philippines

Title of Project: **Contribution to Population Dose from the Transport of Radioactive Materials in the Philippines**

Chief Scientific Investigator: E.M. Valdezco

Other Scientific Staff: V. Kinilitan, B.C. Bernardo (consultant), L. Ilagam

Expiry: 1986 (with possibility for extension into 1987 outside of CRP)

Abstract:

This study was initiated late in the Coordinated Research Programme (CRP), and is expected to continue beyond the termination of the CRP. The study proposes to assess the radiological impact to the population from the transportation of radioactive materials in the Philippines. Actual measurements of radiation dose to transport workers are being undertaken to be correlated with the estimated values that will be derived from INTERTRAN calculations. Results of measurements on transport workers will be extrapolated to account for the total population of the country. To date, the INTERTRAN code has been adopted to the VAX-11/750 computer system with few modifications.

Preliminary results have shown reasonable agreement between measured and calculated values of radiation doses of transport workers. More shipment data and other relevant parameters are being collected to ensure validity of results.

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DETAILED SUMMARY REPORTS

RADIOACTIVE MATERIAL TRANSPORT PACKAGE ACTIVITY LIMITS

RADIOACTIVE MATERIAL TRANSPORT PACKAGE ACTIVITY RELEASE LIMITS*

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1.0 INTRODUCTION

This report presents a summary of work carried out under the terms of Research Agreement No. 2291/CF. Initially this involved an assessment of individual and collective doses arising in the routine transport of spent fuel from commercial nuclear power stations in the UK and an examination of the dosimetric implications of the Type B package activity release limits contained in the 1973 IAEA Transport Regulations. Following on from this, a major component of the work was the development of the Q system, an improved version of the earlier A_1/A_2 system, for the determination of Type A package contents and other limits within the Regulations. Finally, following adoption of the Q system for the evaluation of A_1 and A_2 values appearing in the 1985 revised Regulations, the implications on activity release limits for Type B packages were examined in the context of the transport of irradiated nuclear fuels.

2.0 DOSES ARISING IN THE TRANSPORT OF IRRADIATED NUCLEAR FUELS WITHIN THE UK

Consignments of irradiated fuel from CEGB and SSEB nuclear power stations in transit by road and rail within the UK to the BNFL works at Sellafield, Cumbria, for reprocessing are controlled under the terms of Approvals and Special Arrangements granted by the Department of Transport. This ensures that such transport operations are carried out within the conditions laid down in the IAEA Regulations for the Safe Transport of Radioactive Materials. In order to determine the contribution of these fuel movements to the overall radiological impact of all stages of the nuclear fuel cycle, the individual and collective radiation doses associated with the routine transport of irradiated fuel in the UK were assessed [1,2].

In general, consignments of irradiated fuel in transit may lead to low level exposure of transport workers and the public via several pathways:

(a) External penetrating radiation direct from the fuel flask;
(b) External or internal radiation due to exposure to a release of airborne radioactive material;

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* Research Agreement No. 2291/R3/CF.

(c) External radiation from ground contamination by dry deposition or rainout of airborne radioactivity;
(d) Internal radiation due to consumption of foodstuffs contaminated by deposition of airborne radioactivity.

However, in the case of fuel movements within the UK there are two barriers to the release of airborne activity, namely the flask lid seal and the fuel element cladding for intact fuel or the lid seal and transit bottle for damaged fuel. Decontamination of the external surfaces of the flask prior to dispatch ensures that exposure due to residual levels of surface contamination ($< 10^{-4}$ $\mu Ci.cm^{-2}\beta/\gamma$) is minimal, thus the dominant contribution to doses arises from penetrating radiation emerging direct from the flask and the treatment outlined here concentrates on this exposure pathway.

Measurements of external gamma dose rates around irradiated fuel flasks dispatched from CEGB nuclear power stations indicate a maximum dose rate at 1 m from the flask surface of about 10 $\mu Sv.h^{-1}$ (1 $mrem.h^{-1}$). Further, these observations demonstrate that the flask may be treated as a point source of gamma-rays, leading to relatively simple expressions for the dose rate as a function of distance from static or moving transport flasks [1,2]. This simple model, combined with a knowledge of typical flask movement timetables and assumed exposure patterns for transport workers and members of the public, readily enables the maximum individual doses to various groups to be evaluated, as follows:

(a)	road users/pedestrians	3×10^{-4} mSv
(b)	drivers of road vehicles	0.54 mSv
(c)	workers loading flasks at railheads	0.27 mSv
(d)	public living near transport routes	10^{-4} mSv
(e)	public living near marshalling yards	3×10^{-2} mSv
(f)	rail workers at marshalling yards	9×10^{-2} mSv
(g)	train crews	8×10^{-3} mSv
(h)	rail passengers	10^{-4} mSv

In general, these levels of individual dose are low compared with that due to natural background radiation in the UK, typically 1-2 $mSv.a^{-1}$, and are similar to those derived earlier for Magnox fuel transport [1,2]. Differences arise from changes in the patterns of flask traffic in recent years, which have tended to reduce stopover periods en route to Sellafield, and from a more realistic treatment of exposure of road vehicle drivers and pedestrians during the short journeys from the power station to the nearby railhead.

In order to estimate the population doses resulting from fuel movements associated with CEGB and SSEB reactor operations, the transport routes from the power stations to Sellafield were divided into a series of route elements. Since precise representation of the transport routes would involve a prohibitively large number of route elements, those used in this analysis were selected in order to reproduce the major variations in population density along the actual routes and their locations relative to the main centres of population. The collective dose contributions arising from fuel movements, including stopovers at marshalling yards en route, were evaluated by superimposing the dose distributions for moving and static point sources on the population distribution of the UK as recorded in the 1971 National Census and incorporated in the POPDOS2 code [3]. Flask transit speeds and stopover periods were estimated from current flask movement timetables and lower limits on the range of numerical integrations were applied since restriction of access to railway property which comprises the overwhelming majority of the routes makes it unlikely that the public will closely approach flasks in transit (for details see references 1 and 2).

The annual collective doses to members of the public and transport workers due to Magnox fuel traffic from CEGB and SSEB nuclear power stations calculated as outlined above are as follows:

(a)	road users/pedestrians	0.3 man mSv
(b)	road vehicle drivers	2.7 man mSv
(c)	workers loading flasks at railheads	2.0 man mSv
(d)	public living along transport routes	2.1 man mSv
(e)	public living near marshalling yards	6.1 man mSv
(f)	rail workers at marshalling yards	3.5 man mSv
(g)	train crews	0.05 man mSv
(h)	rail passengers	0.95 man mSv

These estimates make no allowance for the shielding effects of buildings or local topographical features, such as railway embankments or tunnels, which might reduce the contributions to the public. Also, the total population dose of 18 man $mSv.a^{-1}$ associated with Magnox fuel movements is somewhat lower than earlier estimates [1,2] for the reasons cited in the context of individual doses above. This should be regarded as an upper limit value and is associated with an installed capacity of 4.2 GW(e). The comparable figures for the current programme of five AGR power stations, representing an installed capacity of ~ 6 GW(e), and the projected 1.2 GW(e) PWR station at Sizewell B are 8.6 and 0.7 man $mSv.a^{-1}$ respectively.

Overall irradiated fuel movements represent a minor contribution to collective doses arising in the nuclear fuel cycle. The transport contributions derived above represent < 10^{-4}% of the natural background radiation dose to the UK population.

3.0 ACTIVITY RELEASE LIMITS FOR TYPE B PACKAGES

Within the 1973 IAEA Transport Regulations the permitted releases for Type B(U) and Type B(M) packages were specified respectively as $A_2 \times 10^{-3}$ and A_2 over a period of up to one week following a severe transport accident. These are the limits which apply to consignments of irradiated reactor fuel and, for an accident occurring out-of-doors under average weather conditions and exposure within 50-100 m of the flask, can be shown to be equivalent to integrated whole-body dose limits of approximately 10 µSv (1 mrem) and 10 mSv (1 rem) for Type B(U) and Type B(M) packages respectively [1]. These values compare with the Emergency Reference Level (ERL) of dose to the whole body of 10 rem (0.1 Sv) currently used as the basis for the derivation of emergency action levels at CEGB and SSEB nuclear power sites in the UK [4].

Thus the Type B(U) and B(M) activity release limits are respectively four and one orders of magnitude more restrictive than the criterion commonly used for emergency action at power station sites. These differences may be rationalized by noting that transport accidents can occur anywhere along the route from the power station to the reprocessing plant, including relatively densely populated urban areas, whereas power stations tend to be sited in remote areas. Indeed, within the UK the maximum collective dose for a reactor accident at the ERL level is roughly comparable with that for a transport accident occurring in a densely populated urban area with a release corresponding to the Type B(M) limit [1]. Thus, if one associates similar probabilities with these two events, the Type B(M) limits result in similar safety standards for fuel transport and reactor operations. Equally, on this basis the lower Type B(U) limits appear unduly restrictive.

As a result of considerations of the type outlined above, in the 1985 revised Regulations the more restrictive activity release limit for Type B(U) packages has been relaxed such that the earlier Type B(M) limits are now applied to all Type B packages. In addition, explanatory information on the background to the development of activity release limits for Type B packages was prepared for inclusion in the revised edition of Safety Series No. 7.

4.0 DEVELOPMENT OF THE Q SYSTEM

The Q system originated from a critical examination of the dosimetric models used in the derivation of the Type A package contents limits in the 1973 IAEA Transport Regulations [5-8]. It is essentially a development of the A_1/A_2 system incorporated in the 1973 Regulations which takes account of the recent recommendations contained in ICRP Publications 26 and 30. In addition, the radiological protection criteria underlying the derivation of the A_1 and A_2 Type A package contents limits are clearly defined. This is considered to be an important consideration at a time when the nuclear industry is increasingly being called upon to justify its actions in relation to public safety issues.

Under the Q system a number of Type A packge contents limit values, Q_A, Q_B, Q_C, etc., are defined with reference to specific radiation exposure pathways as described below. The A_1 value for special form material is then determined as the least of the two values Q_A and Q_B, while the A_2 value for non-special form material is the least of A_1 and the remaining Q values. A_1 and A_2 values are determined for individual radionuclides and may be applied to known mixtures of different radionuclides or to the derivation of Type B package leakage, exemption of LSA limits as in the 1973 Regulations.

The fundamental assumptions underlying the Q system are:

(a) The effective dose equivalent to a person exposed in the vicinity of a transport package following an accident should not exceed the annual dose limit for radiation workers, namely 50 mSv (5 rem);

(b) The dose equivalents received by individual organs, including the skin, of a person involved in the accident should not exceed 0.5 Sv (50 rem), or in the special case of the lens of the eye 0.15 Sv (15 rem);

(c) An individual is unlikely to remain at 1 m from the package for more than 30 minutes.

Practical considerations make it unlikely that members of the public would receive doses up to the limits cited above. Also, the exposure period of 30 minutes at 1 metre is a cautious judgement of the incidental exposure of persons initially present at the scene of an accident, it being assumed that subsequent recovery operations take place under health physics supervision.

The individual Q values are determined with reference to specific exposure pathways, which are illustrated schematically in Figure 1, as follows:
Q_A - External dose due to photons (gamma or X-rays) assuming complete loss of package shielding. The above dose limits and exposure conditions lead to a radiation dose rate limit from the damaged package essentially the same as that in the 1973 Regulations, i.e., 10 $rem.h^{-1}$ at 1 m $\simeq$ 1 $rem.h^{-1}$ at 3 m.

Q_B - External dose due to beta emitters assuming dose rate variation with maximum beta energy as shown in Figure 2. In addition, shielding factors in the range 2 to > 100, associated with materials such as the source beta window protector, packaging debris, etc., are included in a manner similar to that in the 1973 Regulations.

Q_C - Internal dose due to inhalation of non-special form or dispersible material released from a damaged package. Following discussion at the Special Working Group meeting on the Q system convened by the Agency in London in 1982, the nett intake via inhalation was revised from $6 \times 10^{-6} \times Q_C$ [6] to $10^{-6} \times Q_C$, the same value as in the 1973 Regulations. However, the 10^{-6} intake factor is now based on a combination of a range of respirable aerosol release fractions (10^{-2} - 10^{-3} of the package contents) and a range of uptake factors (10^{-4} - 10^{-3} of the released material).

Q_D - Skin contamination and ingestion doses resulting from handling a damaged package containing non-special form material. 1% of the package contents are assumed to be dispersed over an area of 1 m^2; the hands are assumed to become contaminated to 10% of this level, but to be washed within a period of five hours. The depth of the sensitive basal layer of the skin is taken as 7 $mg.cm^{-2}$, and Q_D values determined using the beta dose rate versus maximum beta energy data shown in Figure 3. Again, following discussions at the Special Working Group meeting an upper limit of 10 $mg.cm^{-2}$ on the mass of material which might be retained on the skin for any significant period is imposed for low specific activity materials. Finally, assuming all the activity from 10 cm^2 of skin is ingested over a 24-hour period it can be shown that almost without exception the external dose to skin is more limiting than that due to either inhalation or ingestion. Also, for maximum beta energies $\leq$ 5 MeV the dose to skin is more limiting than that to the lens of the eye.

Q_E - Submersion dose due to gaseous isotopes assuming a 100% release of the package contents into a storeroom or cargo-handling bay 3 m x 10 m x 10 m with four air changes per hour. This pathway applies to noble gases which do not become incorporated into the body assuming transport in non-special form in a compressed or uncompressed state. Special consideration is given to ^{222}Rn and the lung dose associated with inhalation of its short-lived daughter products using the data recommended in ICRP Publication 32.

Q_F - Contents limit for special form alpha emitters, defined as $10^4 \times Q_C$. Within the Q system there is no dosimetric basis for this arbitrary definition. The increase in the multiplying factor, from 10^3 in the 1973 Regulations to 10^4, was determined by the Special Working Group meeting; this change was justified in part by the ten years of good experience in the transport of special form materials and in part by the reduction by factors of up to 10 in the Q_C values for alpha emitters arising from changes in dosimetric data between ICRP Publications 2 and 30.

Finally, tritium and its compounds are treated as a special case, with elemental tritium being treated as tritiated water for dosimetric purposes. As in the 1973 Regulations an arbitrary upper limit on A_1 and A_2 values of 40 TBq (~ 1000 Ci) is applied and calculated Q values based on a range of accident scenarios lead to results greater than this value for both organic and inorganic forms of tritium. In addition, a specific activity limit for tritium of 1 TBq/l (27 Ci/l) was determined by the Special Working Group.

Overall, the adoption of the Q system introduced only relatively minor changes in the A_1 and A_2 values compared with those in the 1973 Regulations. This is illustrated in Figure 4 for those radionuclides listed in Table IV of the 1982 Edition of Safety Series No. 37. Where major variations occur, say by greater than a factor of 2-3, these are generally associated with reductions in the Type A package contents limits for beta emitters determined by Q_B and particularly Q_D. This latter Q value refers to a dosimetric route not previously considered in the Regulations, namely beta contamination of the skin of exposed persons. Isolated relaxations in contents limits occur due to changes in dosimetric data introduced in ICRP Publication 30, as in the case of ^{90}Sr, and more generally in the case of special form alpha emitters due to the revised defintion of Q_F.

The A_1 and A_2 values evaluated using the Q system for inclusion in the 1985 revision of the Regulations were verified by Dr. Keith Eckerman of Oak Ridge National Laboratory under the sponsorship of the US Department of Transportation. In addition, ALI values for radionuclides not inluded in ICRP Publication 30 were supplied by Mr. Ken Shaw of the UK National Radiological Protection Board. Finally, under the terms of this Research Agreement a report containing a comprehensive description of the Q system was prepared [9], which in large measure forms the basis for Appendix I in the revised edition of the Safety Series No. 7.

5.0 RADIOLOGICAL LIMITS IN THE TRANSPORT OF IRRADIATION NUCLEAR FUELS

The final stage of the work under the terms of this Research Agreement comprised a review of the radiation levels and activity release limits applicable to irradiated fuel transport flasks, particularly with regard to changes introduced in the 1985 Regulations [10]. The various provisions of the Regulations for normal transport and under accident conditions were expressed as equivalent dose rate and dose limits under a range of potential exposure situations.

Using isotopic inventories calculated for a representative range of thermal reactor fuels it was shown that the adoption of the Q system has led to a relaxation in the permitted activity releases from Type B packages by a factor of about 2 compared with the earlier 1973 Regulations. The A_2 values for known mixtures of radionuclides were evaluated according to the expression given in the Regulations and used to express the Type B limits in terms of the amounts of fuel material which could be involved. For example, in the case of PWR fuel irradiated to 35 $GWd.t^{-1}$ at 35 $MW.t^{-1}$ and cooled for 100 days the Type B limits derived in this manner are 0.65 µg of fuel material per hour and 0.65 g of fuel over a period of up to one week for normal transport and severe accident conditions respectively. These permitted release rates represent the amount of material which may be released from the flask, rather than the total fuel involvement which could be somewhat greater when allowance is made for the respirable aerosol fraction and any hold-up mechanism resulting in retention within the flask of a proportion of the material released from damaged fuel elements.

The limiting dose rates and doses implied by the various provisions of the Regulations were also derived, noting that an intake of $A_2 \times 10^{-6}$ corresponds to an effective dose equivalent of 50 mSv (5 rem). Under normal transport conditions the activity release and external radiation limits specified lead to comparable limiting dose rates at distances likely to be of interest in irradiated fuel transport operations, namely a few $\mu Sv.h^{-1}$ within a few metres of a Type B package being handled indoors or $\sim 10^{-2}$ $\mu Sv.h^{-1}$ at a distance of about 100 m for a package in transit out-of-doors. For exposure indoors a Type B package leaking at a rate of $A_2 \times 10^{-6}$ per hour in a large storeroom or cargo handling bay of typical dimensions 30 x 10 x 10 m^3 with 4 room air changes per hour was considered. Under equilibrium conditions and assuming uniform mixing this leads to an intake rate of about $A_2 \times 10^{-10}$ per hour, representing a dose rate limit of 5 $\mu Sv.h^{-1}$ (0.5 $mrem.h^{-1}$). This may be compared with the external radiation limit of 0.1 $mSv.h^{-1}$ (10 $mrem.h^{-1}$) at 1 metre from the surface of a Type B package or at 2 metres from a package assembled as a full load, which in turn corresponds to a dose rate limit of order 1 $\mu Sv.h^{-1}$

($0.1\ mrem.h^{-1}$) at distances in the range 10–20 metres. The equivalent situation for exposure out-of-doors under average weather conditions is illustrated in Figure 5, while the variation of dose with distance from the source for an accidental release of A_2 over a period of a few hours is shown in Figure 6. For longer duration releases the doses would be reduced compared with those shown in Figure 6 due to the increased lateral spreading of the plume. Thus under severe accident conditions the release of A_2 in a period of up to one week corresponds to an effective dose equivalent limit up to ~ 10 mSv (1 rem) for exposure out-of-doors at distances in the range 50–200 m from the source. This level of dose is at least an order of magnitude greater than the corresponding dose over a period of one week implied by the external radiation limits under accident conditions specified in the Regulations. Exposure indoors following an accident was not considered since it is unlikely that accidents of the severity which is simulated in the Type B tests specified in the Regulations would occur indoors, or if they did the resulting conditions would be such as to require immediate evacuation of all persons in the vicinity [8].

Although the radiation doses received in any transport operation or accident situation may vary from the values cited above, depending on the detailed release mechanisms and isotopes involved, the study outlined here indicates that the levels of dose involved are low. In the case of normal operation the implied dose rate limits close to a Type B package are comparable with natural background radiation dose rates, while under severe accident conditions the implied dose limits are within those widely used in establishing emergency action levels at power station sites (as discussed earlier).

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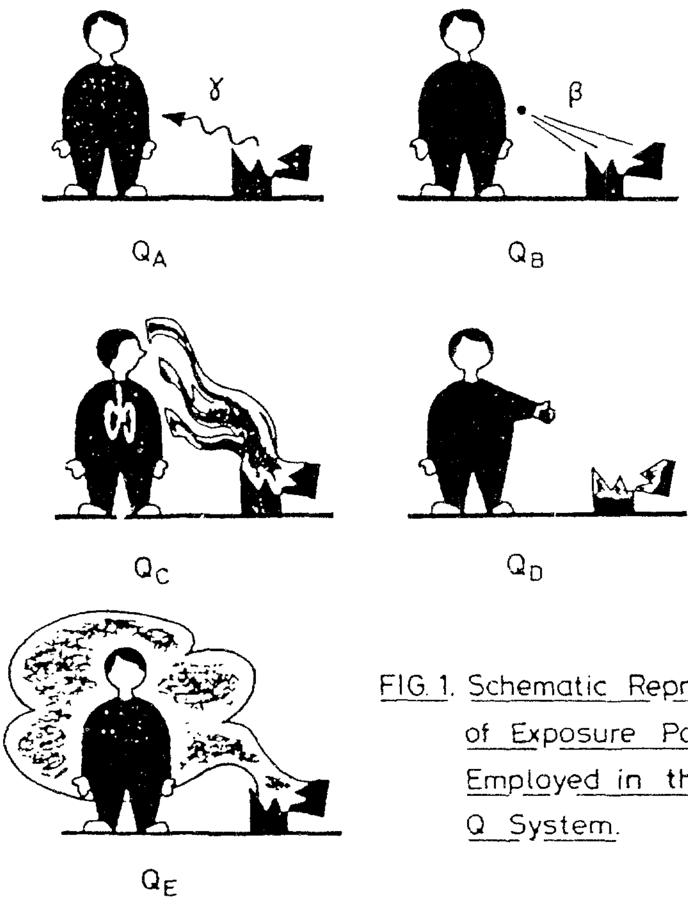


FIG. 1. Schematic Representation of Exposure Pathways Employed in the Q System.

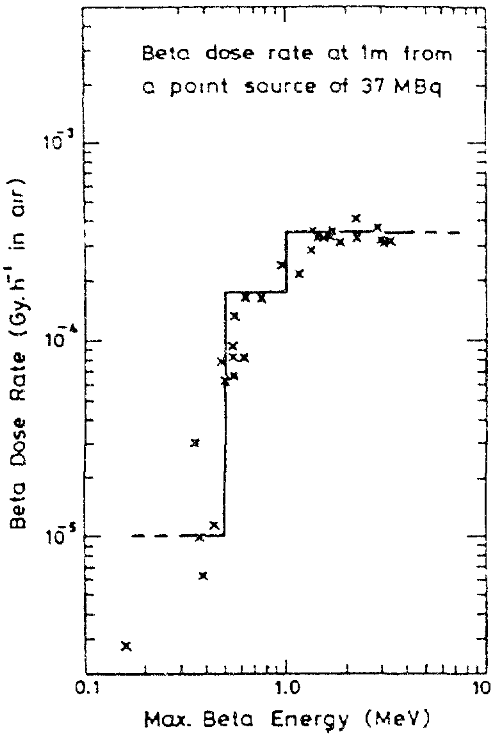


FIG.2. Variation of External Beta Dose Rate to Skin with Energy.

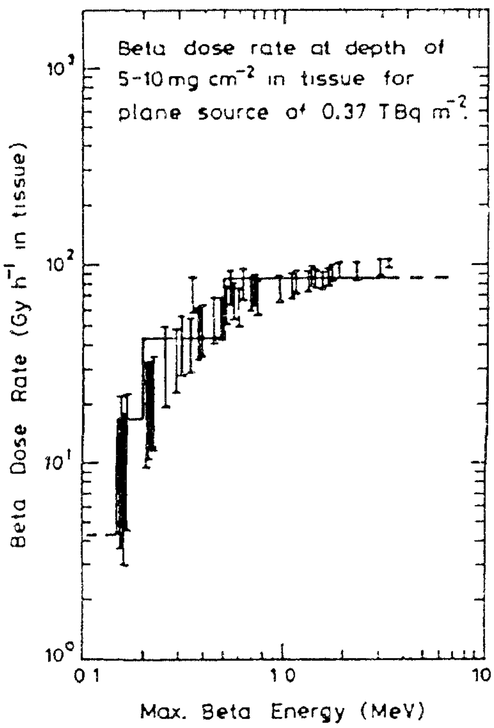


FIG 3 Variation of Beta Dose Rate Due to Skin Contamination with Energy.

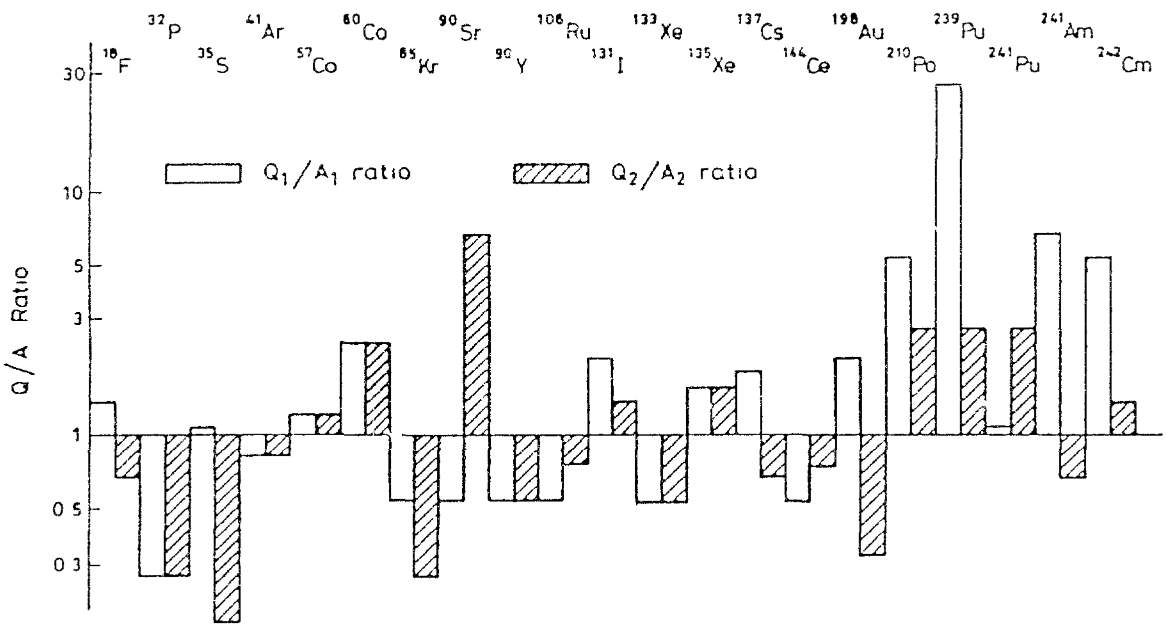
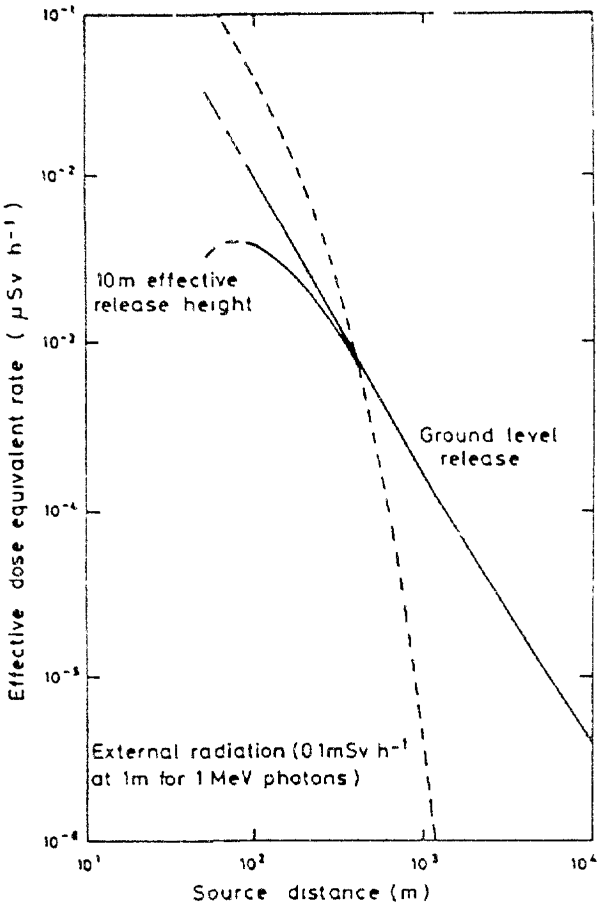
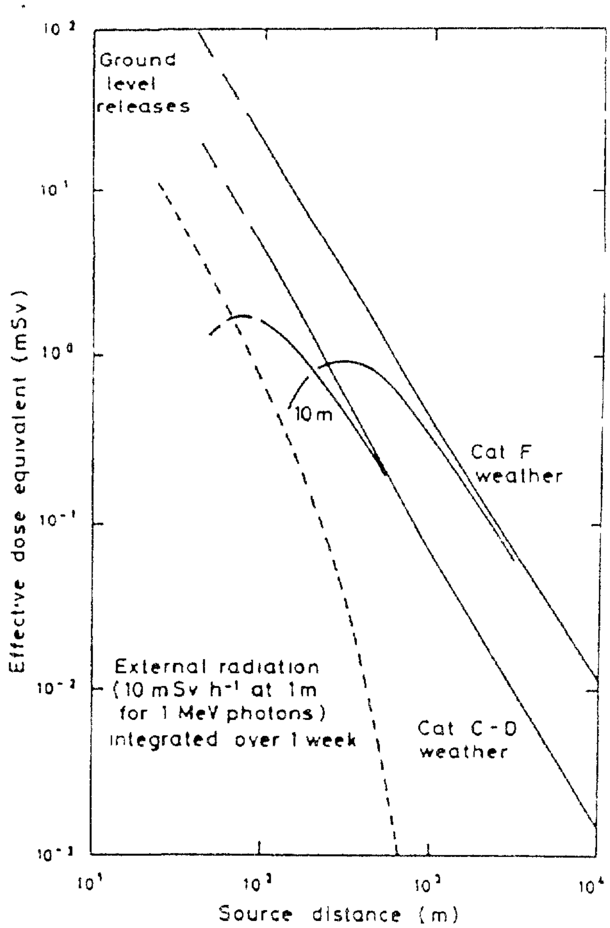


FIG.4 Comparison of Type A Package Contents Limits Evaluated Using the Q System (Q_1/Q_2 values) with Those in the 1973 IAEA Transport Regulations (A_1/A_2 values)



Normal Transport Activity Release and Radiation Limits Expressed as Dose Rates Under Average Category C-D Weather Conditions.

FIG.5.



Accident Activity Release and Radiation Limits Expressed as Integrated Doses Under Various Conditions

FIG 6

PACKAGE/CONTENTS BEHAVIOUR IN NORMAL AND ACCIDENT ENVIRONMENTS

CANDU IRRADIATED FUEL TRANSPORTATION: THE SHOCK AND VIBRATION PROGRAM*

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1.0 INTRODUCTION

As part of the irradiated fuel transportation project initiated in 1975, a shock and vibration program was set up to evaluate the response of the irradiated fuel to the transportation environment. Since that time, shock and vibration levels and fuel bundle response have been characterized through:

(a) Road transportation field measurements of normal vibration and transients;

(b) Rail transportation field measurements of normal vibration, transients, and rail car coupling impact;

(c) Modal testing of the fuel transport containers;

(d) Impact and fatigue testing on irradiated fuel and unirradiated fuel;

(e) Analytical modelling of the road and rail transportation modes.

The primary objective of the shock and vibration program was to show how the fuel bundles and containers (modules) can be transported without failure. Within the context of the shock and vibration program a fuel bundle failure is defined as disassembly or distortion prohibiting normal automated fuel handling. This is a major consideration for CANDU fuel because of the large number of bundles that must be handled.

Over the years changes in the shock and vibration program have occurred in response to cask design changes which were identified in part by the program itself. Consequently, there were two phases to the program. Both phases combined a substantial amount of experimental and analytical work.

The main differences between Phase 1 and Phase 2 arose out of the cask concept selection process narrowing down to a single concept from a host of others. During Phase 1 it was assumed that individual bundle restraints and module stack restraints would be needed. Consequently, impact and fatigue testing of bundles in Phase 1 was conducted using an axial preload on the bundles.

The results of these impact and fatigue tests proved the bundles to be highly resistant to failure. Therefore, in Phase 2, the impact and fatigue testing was repeated to establish threshold levels for axially unrestrained bundles.

For the analytical modelling, the main differences between Phase 1 and Phase 2 again relate to having a more specific road cask design in mind. In addition, some of the modelling methods were improved.

* Research Agreement No. 2715/R1/CF.

2.0 THE CANDU FUEL BUNDLE, MODULE CONTAINER AND CASK CONCEPTS

2.1 Fuel Bundles

The CANDU fuel bundle used in all of Ontario Hydro's major nuclear power generating stations is made up of either 28 or 37 "fuel elements" held together in a fixed array. The overall bundle diameter is about 10 cm and its length, about half a metre. Figure 1 illustrates a 37 element bundle, which is used at the Bruce stations, and will be used at Darlington. The 28 element bundles are used at Pickering. The overall dimensions of the two bundle types are the same, although their masses are slightly different.

2.2 The Module

The module is both an irradiated fuel storage container and a shipping container. Figure 2 illustrates the module. It is made of 304 stainless steel. Two tube sheets, 4.75 mm thick and spaced about 575 mm apart, hold the 48 tubes in place. The tubes are very nearly one metre in length, 105 mm inner diameter and 1.5 mm thick. As bundles are handled in pairs, one module holds 96 fuel bundles. The fuel is horizontal in storage as well as transport.

2.3 Cask Concepts

During Phase 1 of the shock and vibration program a variety of cask concepts were evaluated. The major variables were road casks and rail casks, and cylindrical geometries and rectangular geometries. Because of the variable spaces between the module and the inner wall of a cylindrical cask, bundle restraint mechanisms would have been necessary. Even for the rectangular cask concepts, restraint mechanisms were assumed to be required to provide a preload. However, Phase 1 results showed the bundles to be highly tolerant of impact and fatigue loads. With a desire to reduce complexity in design and operation, it was decided in 1983 to develop only a rectangular cask with no bundle restraint.

Under the Phase 2 assessments no bundle restraint is assumed. Axially, movement of the bundles is limited only by friction, while travel is limited by the cask walls. The total amount of travel permitted assumed in Phase 2 is 33 mm based on a rectangular cask design. Lateral and vertical travel of the bundle is limited by the module tube.

3.0 PHYSICAL TESTING OF SYSTEMS AND COMPONENTS

3.1 Road Field Tests [1]

Road field tests were done using a tractor-trailer loaded with a module placed inside a concrete mass, simulating a 32 tonne cask. Accelerometers were mounted on the tractor drive axle, trailer axle, baseplate on which the module was placed (horizontal and vertical components) and on a module tube. This vehicle was driven a total of 900 km over a variety of road conditions. The tractor unit was a tandem drive with steel spring suspension. The trailer was a single drop deck triaxle with steel springs. The trailer mass was 9.1 tonne and the combined tractor-trailer mass was 18 tonne. Table 3-1 summarizes the overall steady-state accelerations (vertical component only which is generally the largest component) for the truck tests.

Power spectral densities show that most of the steady-state vibration energy is in the 0 to 300 Hz range. The significant peaks are at about 2 Hz and at about 15 Hz. The vibration of 2 Hz corresponds to the trailer's first natural frequency.

The response of several road transients was also recorded. Data for the truck crossing railway tracks were analyzed. At the significant frequency of 2 Hz the response at the baseplate is higher than normal levels by a factor of about 1.4. At the other significant frequency, about 15 Hz, the baseplate resonse is 2.8 times higher than corresponding normal levels. Further analysis of the same transient showed amplification factors between the baseplate and central module tube response to be 1.1 at 2 Hz and 3.4 at about 15 Hz. Significant amplification occurs at higher frequencies but are of little consequence as the vibration energy transmitted is small.

3.2 Rail Field Tests [1]

A rail flat car carrying a module in a simulated 68 tonne cask was instrumented with accelerometers and hauled a distance of 420 km over a variety of track conditions. The rail car was a Canadian National series 667 flat car, of 91 tonne capacity, equipped with standard draft gear couplers. The mass of the flat car was 28.9 tonne. Accelerometers were mounted on the rail car rear bogie frame, the baseplate (horizontal and vertical components) and some module tubes. Table 3-1 also summarizes the overall steady-state vibration levels (vertical component).

TABLE 3-1
ROAD AND RAIL NORMAL VERTICAL ACCELERATION LEVELS

		Acceleration (Vertical in g's (peak))		
Mode	Location	Mean	Min. (peak)	Max. (peak)
Road	baseplate	0.10	0.06	0.18
	central module tube	0.30	0.15	0.45
Rail	baseplate	0.14	0.10	0.20
	central module tube	0.20	0.16	0.30

Power spectral density at the baseplate show most of the vibrational energy occurs over the 0 to 500 Hz range. The significant responses were at 1.75 Hz, 15 Hz, 22.5 Hz, 90 Hz, 232.5 Hz and 252.5 Hz. The 1.75 Hz response corresponds to the first natural frequency of the rail car.

Transients under normal rail transport, which include the taking up (and giving out) of slack in the couplers and track irregularities such as at switches, were measured. Most of the energy is taken up at low frequency, near the rail car's natural frequency. The transient response is about ten times that of the normal vibrational response. However, because a CANDU fuel bundle's lowest natural frequency (about 35 Hz) is above the response range (0 to 25 Hz) where most energy is absorbed, little detrimental effect is expected.

Rail car coupling impact tests were also conducted, at National Research Council facilities, Uplands, Ontario. However, cask response results were of limited value due to slippage of the concrete blocks used to simulate the cask mass, but tend to confirm g-loads from the literature and analyses.

3.3 Fuel Bundle Fatigue and Impact Tests [2,3]

A series of tests have been done to establish endurance limits for fuel bundles under fatigue vibrations and impact shocks. These limits can be compared with field test acceleration levels and the analytical model results of the transport environments. Test fixtures were designed to house two fuel bundles for the fatigue tests and lateral and axial impact tests. The fixtures consisted of a single tube fixed between two sheets, duplicating the support structure of the module container. The single tube arrangement was fixed within a steel casing. For Phase 1, the mechanisms provided a means of applying the 6 g axial preload. For Phase 2 the mechanisms provided the 33 mm axial clearance.

A 24 h period was selected as the reference time interval for the fatigue testing as this period would likely encompass the travel time to any off-site facility in Ontario. Various levels of acceleration could be applied. The frequency of vibration was 21 Hz, selected because it avoided resonances in the test apparatus, and was close to the major responses measured from the field tests. Peak acceleration levels ranged from 0.5 g to 3.0 g. In some trials bundles were vibrated at low levels for a 24 hour period, then raised to higher acceleration levels for about one hour. This is assumed to conservatively cover transient acceleration loads in addition to normal vibration.

Two bundles were fatigue tested simultaneously at acceleration levels starting at 0.5 g (peak). If at the end of the 24-hour period failure did not occur, the level was raised up to 3.0 g (peak) for up to one hour duration. This is assumed to conservatively cover transient loads in addition to normal vibration. The next pair of bundles was then tested for 24 hours at a higher acceleration level until a failure threshold level was established. Results of the fatigue vibration thresholds established are summarized in Table 3-2.

For impact testing, the test fixtures were suitably "cushioned" to provide response rise times of 5 ms to 10 ms. This was based on rise times reported in the literature for rail car coupling impacts. Rail car coupling is the worst shock environment the bundles will experience under normal transportation.

The Phase 1 impact tests produced no bundle failures. In fact, no cracks were induced. The highest deceleration levels achieved in the axial and lateral orientations are summarized in Table 3-3. Constraints within the test apparatus in both phases did not permit impact testing to failure levels. It should also be noted that bundles were impacted several times each, beginning at lower levels and working up to the maximum levels achievable. Furthermore, some of the impacted bundles had previously been fatigue tested for a period of 24 hours or more.

The impact test results from Phase 1 proved the bundles to be highly tolerant to impact. The second phase repeated similar fatigue and impact tests with a 33 mm axial clearance. In the Phase 2 axial impact tests three clearance modes were tested. Mode 1 has all the clearance in between the two bundles. Mode 2 has all the clearance at the front of the bundle pair. Mode 3 has half the clearance at the front and half the clearance in between the two bundles. An extra test was also performed on a bundle pair which had been previously subjected to a fatigue test for 24 hours at 7.35 m/s^2 (0.75 g) peak. This was done in order to determine if the combination of tests would induce a bundle failure.

Although cracks occurred in some endplates, all bundles remained integral and no failure occurred, even for the two bundles that had been previously fatigue tested. A typical time history is shown in Figure 3. As can be noted from the graph, the initial deceleration peak is followed by a secondary peak of the opposite sign (rebound). The impacts occur when the bundles take up the axial clearance in the test fixture. (The initial impact corresponds to the deceleration of the fuel module centre of gravity in the analytical modelling.)

The secondary impact is always substantially greater than the initial impact. The maximum values obtained for initial and secondary peaks were $-390\ m/s^2$ (-39.8 g) peak and $+ 1285\ m/s^2$ ($+131.0$ g) peak respectively, and occurred on the same test. The second worst damaging mode was Mode 3; Mode 1 was the least damaging. The highest accelerations from the Mode 2 configuration are reported in Table 3-3.

The bundles, which had previously undergone fatigue tests at the highest level for 24 hours were also tested in the Mode 2 configuration. Neither bundle failed after exposure to the combined fatigue and axial impact maximum levels.

TABLE 3-2

ESTABLISHED THRESHOLD FATIGUE LEVELS AFTER A 24 h PERIOD AT 21.25 Hz

	Bundle Type	
	28-Element	37-Element
Phase 1 axial restraint with a 6 g preload	0.75 g peak	0.50 g peak(a)
Phase 2, no restraint and 30 mm clearance	N.T.(b)	0.7 g peak

(a) Next highest acceleration level tested was 0.8 g.

(b) N.T. - Not Tested: Only 37-element fuel was tested as 28-element fuel proved to be more robust in Phase 1.

TABLE 3-3
MAXIMUM DECELERATIONS IN IMPACT TESTS

Conditions	Bundle Type (Elements)	Lateral Impact -g	Lateral Impact Risetime (ms)	Axial Impact -g	Axial Impact Risetime (ms)
Phase 1, Axial restraint with a 6 g preload	28	55	6.0	165	5.0
	37	78	5.0	175	5.5
Phase 2, no restraint and 33 mm clearance	37	40	5-10	36 -113	9.2(a) 2(b)

(a) Initial impact, corresponding to deceleration of module, Mode 2.

(b) Secondary impact, bundles taking up clearance.

3.4 Dynamic Testing of a Two Module Stack [4]

For Phase 2, the reference cask capacity (for a road cask) was known to be two modules (192 bundles). To assist analytical modelling of the module/bundle response, modal analysis of a two module stack was done. Two modules were completely loaded with unused fuel bundles. The bundles were not restrained axially. An impact method was used to determine the transfer function that characterizes the vibration transmission through the stack. Resonances occurring between 0 and 1000 Hz were observed. There were ten resonances in all, beginning at 196.8 Hz.

4.0 ANALYTICAL MODELLING OF THE TRANSPORTATION ENVIRONMENT

4.1 Normal Road Transport

Phase 1 [5]

The objective of the normal road transportation modelling in Phase 1 was to determine system response to and accelerations in various road conditions. A two dimensional model was developed, modelling the vehicle and cask as beam elements, which included six degrees of freedom. The input was assumed to be a stationary random process. Power spectral density plots were calculated for the vehicle components. From these, RMS responses of the cask centre of gravity were calculated.

Phase 2 [4]

In Phase 2, modelling of the cask/vehicle system was improved. Results of the modal analysis of the two module stack were used to predict module tube responses. A further improvement was modelling the cask as plate elements, considering in-plane forces and out-of-plane bending, and including the flexibility of the tiedowns. Several response power spectral densities were calculated, and RMS acceleration determined at various positions. The accelerations calculated are higher than those calculated in Phase 1, in general by about 60 percent, and are in better agreement with results of the road field tests. This is to be expected, largely due to the improvements in cask modelling. Table 4-1 summarizes some of the results for the top row of module tubes. Acceleration levels at tubes of the bottom row of the top module are about half of those reported in Table 4-1. The module stack was not restrained vertically.

TABLE 4-1

ACCELERATION AT TOP ROW OF MODULE FOR VEHICLE SPEED OF 80 km/h

Road Condition	Good	Average	Poor	Very Poor
Acceleration (g RMS)	0.06	0.14	0.36	0.73

These results can be compared to the results of the fatigue tests, where the peak acceleration threshold (as an input to the module tube) was established as 0.75 g, or about 0.53 g RMS over a 24 hour period. Therefore, extended travel on very poor roads should be avoided.

4.2 Normal Rail Transport [6]

Phase 1

The objective of the normal rail transport modelling was to determine system response to periodic and random inputs from the tracks. The input function is based on Reference 7. A two dimensional model was used based on the developments of Samaha [8]. Several important non-linear properties are accounted for in the model.

Vertical displacements and rocking responses as a function of train speed were determined. Peak rocking responses occurred at a speed of about 16 to 22 km/h. Peak displacement (vertical) responses occurred at speeds of 28 to 35 km/h. The maximum vertical acceleration was 0.32 g RMS.

Some sensitivity analysis on the rocking response was done by varying damping and suspension spring stiffness. In general, rocking is reduced by increasing the damping or decreasing spring stiffness.

Phase 2

Normal rail transport was not remodelled during Phase 2. The road field tests showed the response at the module tube to be higher for road transport over rail. It can also be reasonably concluded that vibration levels would be below the fatigue thresholds established. For example, if model improvements again raised the calculated response by 60 percent (as they did for the road case), then the peak vertical response for rail transport would rise from 0.32 g RMS to 0.5 g RMS. Fatigue threshold levels for a 24 hour period are 0.53 g RMS.

4.3 Rail Car Coupling Impact

Phase 1 [9]

The objective of the rail car coupling impact modelling was to determine system responses to the event. A two dimensional model was developed, treating the rail car and cask as beam elements. The mass of the anvil car was 30 tonne and the cask was assumed to be 68 tonne. This anvil car rolls into a stationary train at various speeds up to 16 km/h. The mass of the stationary train was 227 tonne.

From the free vibrational analysis, the first ten natural frequencies of the system were determined, and used in the solution of the transient analysis. The ten natural frequencies ranged from 2.3 Hz to 305.2 Hz. The first natural frequency compares well enough with that found in the rail car field tests, 1.75 Hz. To solve th transient equation of motion a numerical method known as the "Newmark-β method" was used.

Three types of coupling draft gears were modelled:

(a) Friction
(b) Hydraulic
(c) Friction-Hydraulic.

Table 4-2 summarizes the acceleration calculated at various positions for initial impact speeds of 16 km/h. It is assumed that travel of the gear is adequate to prevent non-linear behaviour.

TABLE 4-2

PEAK ACCELERATIONS FROM 16 kPh IMPACT

	Friction	Hydraulic	Friction-Hydraulic
Horizontal deceleration of rail car at point of impact	49 g	37 g	53 g
Horizontal acceleration of cask centre of gravity	47 g	36 g	51 g
Vertical acceleration of cask centre of gravity	11 g	9 g	12 g

The effects of limited stroke in the coupler was also determined. For a friction-hydraulic gear with a standard stroke limited to 89 mm, instead of 254 mm, and an impact speed of 16 km/h, the deceleration rises to 475 g from 53 g. At lower impact speeds of 5 km/h the peak deceleration for the same coupler rises from 21 g to 49 g.

Another parameter varied in the analysis was the tiedown stiffness. Over two orders of magnitude variation in tiedown stiffness has a small influence in acceleration responses.

Phase 2 [4]

For the Phase 2 modelling of rail car coupling impact a few changes from Phase 1 were made. The main changes were modelling the two module road cask and its tiedown from totalling 38 tonne, and solving the transient equations of motion using a modified fourth order Runge-Kutta method. The use of a hydraulic and a hydraulic-friction draft gear with "unlimited" stroke are assumed. The simulation has the car with the cask rolling into a stationary train, of 227 tonne, at speeds up to 16 km/h.

The accelerations with time, horizontally and vertically, of the centre of gravity of the two module stack is illustrated in Figures 4 and 5 for the hydraulic-friction coupler and an initial impact velocity of 16 km/h. Peak levels calculated are 10 g horizontally and 3.5 g vertically. The module stack is unrestrained vertically.

These accelerations time histories were transformed into horizontal and vertical vibration frequency profiles. These profiles are then used as an input to the module stack, at its base. Using the data from the module stack modal testing, the response at the module tubes is calculated.

Figure 6 shows, for the top row of module tubes, that high vertical acceleration responses of about 70 g occur at high frequencies. The predicted displacements associated with such accelerations at the frequencies are less than 0.1 of a millimetre. The accelerations at the bottom row of module tubes of the upper module are about half those of the top tubes.

5.0 CONCLUSIONS

(a) CANDU fuel bundles are highly resistant to impact shocks. Unrestrained pairs of bundles can withstand, in the least, lateral impacts of 40 g and axial impacts of 36 g.

(b) CANDU fuel bundles are adequately resistant to fatigue cracking leading to bundle disassembly. Established thresholds of 0.75 g over a 24 h period (0.53 g RMS) are greater than measured and calculated vibration levels for normal road and rail transportation, with one exception. Calculations for very poor roads show predicted vibration levels that could exceed the bundle threshold. Extended travel on very poor roads is to be avoided.

(c) The CANDU fuel bundles need not be axially restrained wtih the cask cavity during road or rail transport. The two module stack within the cask cavity need not be restrained vertically. Bundle and module failures preventing automated fuel handling are not expected.

(d) Rail cars carrying a fuel cask should have a coupler with a long stroke of at least 255 mm. Hydraulic type draft gears produce lower acceleration levels over friction types.

(e) The effective stiffness of the tiedown system, over a 10^7 to 10^9 N/m range, has little effect on the response of the system during rail car coupling impact.

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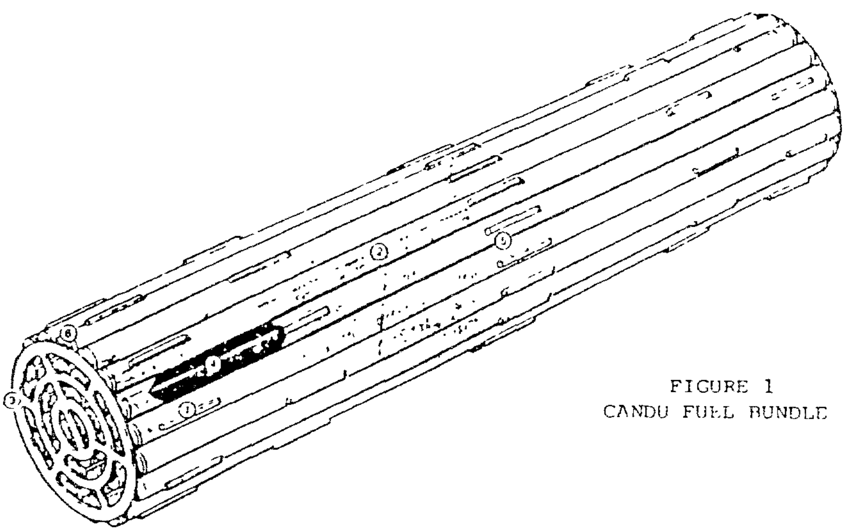

FIGURE 1
CANDU FUEL BUNDLE

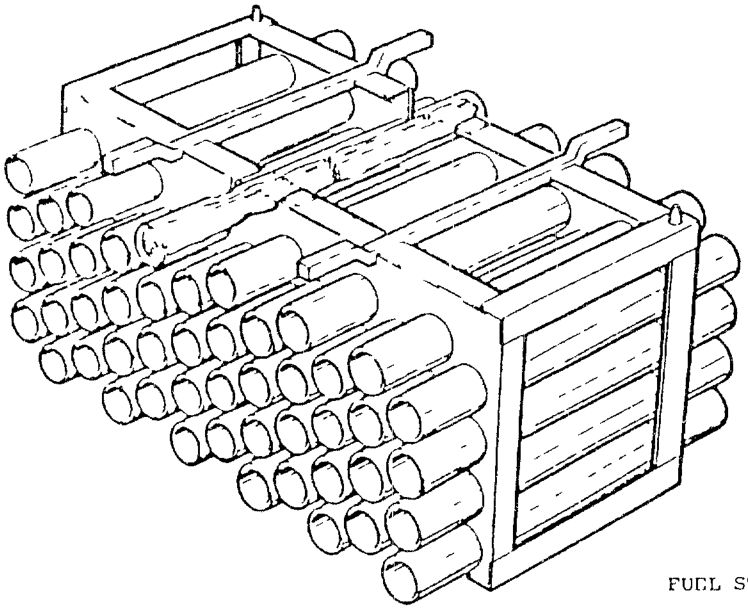

FIGURE 2
FUEL STORAGE AND SHIPPING MODULE

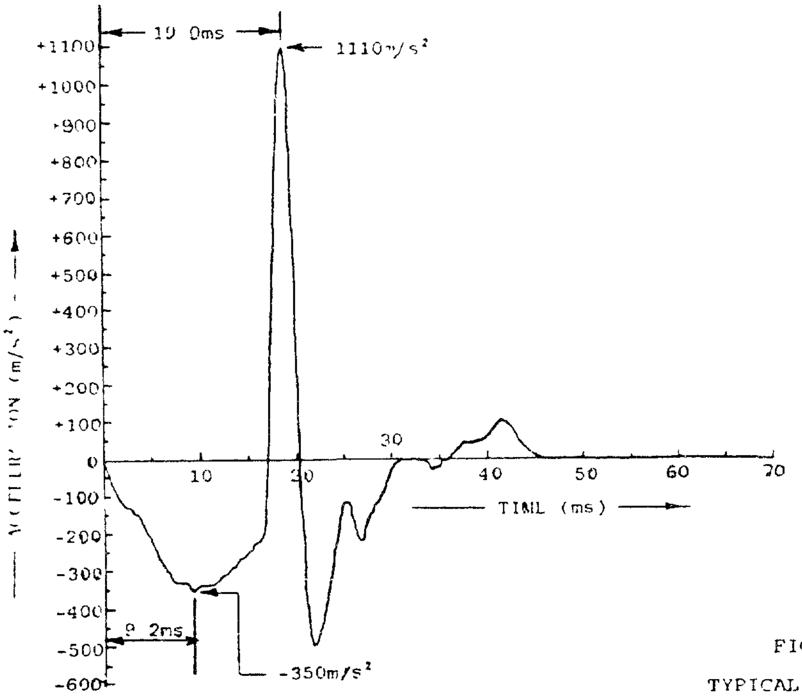

FIGURE 3
TYPICAL TIME HISTORY
FOR AXIAL IMPACT

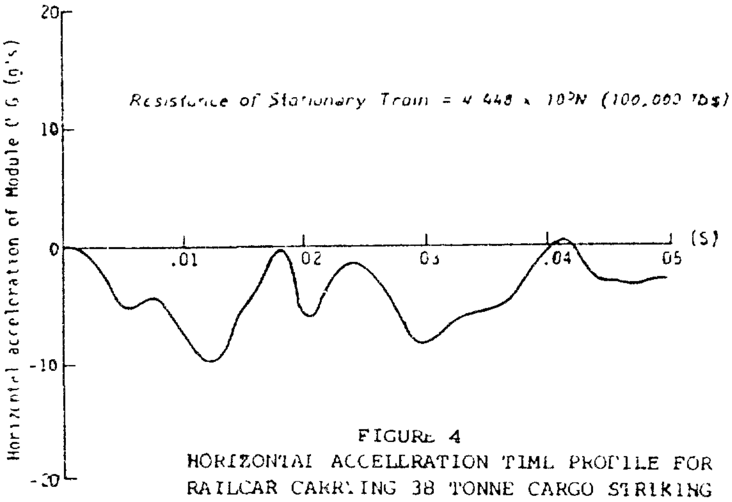

FIGURE 4
HORIZONTAL ACCELERATION TIME PROFILE FOR
RAILCAR CARRYING 38 TONNE CARGO STRIKING
226.8 TONNE STATIONARY TRAIN AT 16KM/H

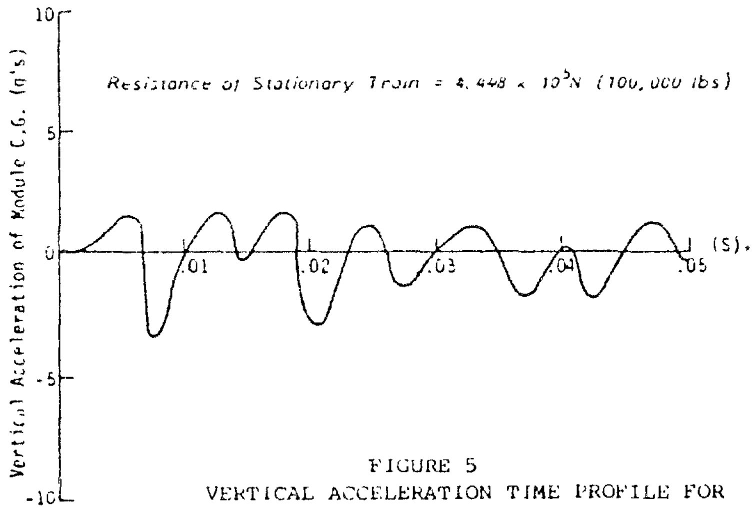

FIGURE 5
VERTICAL ACCELERATION TIME PROFILE FOR RAILCAR CARRYING 38 TONNE CARGO STRIKING 226.8 TONNE STATIONARY TRAIN AT 16KM/H

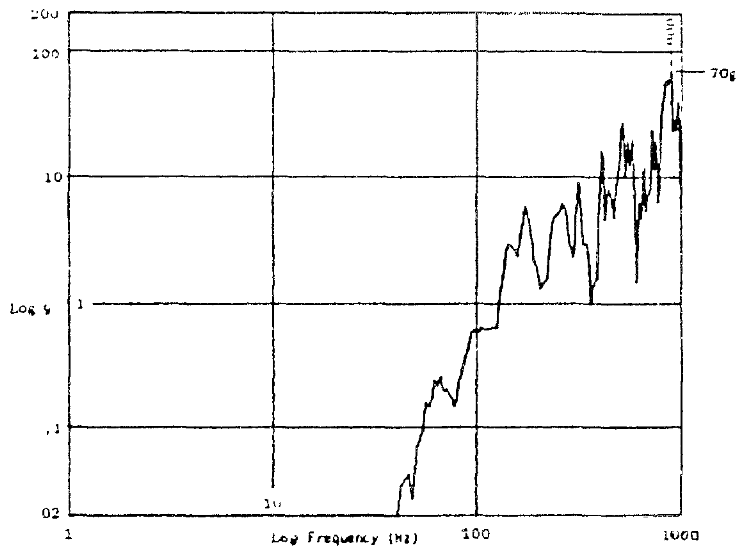

FIGURE 6
VERTICAL SHOCK REPONSE OF AN UPPER MODULE TUBE

EVALUATION OF THE SAFETY MARGINS FOR THE RESISTANCE OF RADIOACTIVE MATERIAL TRANSPORT CONTAINERS DURING A FIRE OF LONG DURATION*

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1.0 INTRODUCTION

The international regulations concerning the transporation of radioactive material are assumed to provide a high level of safety. This is mainly obtained by adapting the design of the package to the potential risk. For exempted, industrial and Type A packagings, their integrity in the case of accidents is not imposed, because the consequences are negligible. For material of higher potential risk, the packagings must resist severe accidental conditions, defined by the regular Type B tests (fire, mechanical and immersion). Nevertheless we have to consider that some accident environments will be more severe than the criteria and, taking into account the traffic involved, the probability to suffer more severe conditions may be of the order of magnitude of the accidents which are considered for the design of nuclear plants. Happily many of the radioactive material packagings which are commonly used are so designed that their integrity is guaranteed in environmental conditions much more severe than the Type B accident conditions.

From these considerations, any risk evaluation must be done with as realistic as possible values, concerning the behaviour of packagings, function of the type of accident (fire, shock, immersion) and the resulting radiological source term.

This is a fundamental objective, to obtain the results of realistic result with risk evaluation codes like INTERTRAN.

Another important point is that risk evaluation combined with methods like cost-benefit, or multicriteria analysis is the tool which permits the determination of options which are the most advantageous to decrease the risk if it is considered as unacceptable and by describing realistic scenarios to establish the most efficient emergency plans.

For all these reasons, it is necessary to develop a data bank of the behaviour of the main packages which are commonly used.

One important case to study is the behaviour of packages under fire conditions - for two main reasons - the probability of fire occurrence having duration and/or temperatures higher than Type B conditions (30 minutes, 800°C) is not negligible for two modes of transport (air and sea), and the source term which often is airborne and consequently offers an inhalation risk.

This is a progress report on the work carried out by the Commissariat à l'Energie Atomique, sponsored by the Institut de Protection et de Sûreté Nucléaire, on the evaluation of safety margins of certain types of containers

* Research Agreement No. 2792/R2/CF.

under fire environment accident conditions. Four different assessments of Type B packagings have been performed as follows:

1) An 800°C fire test was performed on the SV16, a small flask used for transporting Co-60 sources. The flask had previously been crushed and pierced at four points.

2) Thermal analyses of an IL46 flask, used for transporting irradiated fuel elements, were performed for various package configurations and fire durations.

3) The thermal response of a TN-12 irradiated fuel flask, developed in France by TRANSNUCLEAIRE, was analyzed for fire exposures of 30, 60 and 90 minutes duration.

4) The French container FS47, used for the transport of plutonium oxide (PuO_2) was analyzed for fire exposures of 30, 60 and 90 minute duration and fire temperatures ranging from 800°C to 1000°C.

In addition, assessments of two packagings not designed specifically to withstand accident conditions have also been made.

1) The 48Y packing used for shipping natural UF_6 is of the industrial type (material with low specific activity). The regulations do not require a guarantee of fire-resistance for this type of packing. Subsequent to the Mont-Louis accident, the decision was made to reevaluate the safety margins provided by these containers using more realistic assumptions and covering a wider range of potential fires (200°C to 800°C, 30 min. to 48 hours).

2) The use of Type A packing, where the contents are not in a special form, is subject to restrictions when carried through long tunnels (the Mont Blanc Tunnel between France and Italy and the Fréjus Tunnel between France and Italy); this is why the total activity carried per vehicle is limited to 1/3 A_2. A major part of radioactive material is transported from France through these tunnels, especially "ELUMATIC III" packages containing Mo-99 sources used as a technetium 99m generator. The new regulations have restricted the Mo-99 limit by a factor of 10 (10 curies instead of 100). This has led to significant increases in shipping costs. An evaluation of the risks involved is underway so as to reevaluate these restrictions for tunnels. During this study, tests have been carried out on the behaviour of these generators in case of fire to determine the source term for the environment after container integrity loss.

The following briefly summarizes the results from these six separate investigations.

2.0 EVALUATION OF SAFETY MARGINS OF TYPE B PACKAGES

Type B packages are designed to survive a thermal exposure of 800°C for 30 minutes. The following assessments were made of the safety margins of four Type B packages relative to this design criteria.

2.1 SV16 Thermal Test

The mechanical tests which preceded the thermal tests were more severe than is required by regulation. Specifically, having been crushed on two faces and then pierced as far as its central lead shield by two square bars, the SV16 flask was in the condition of a structure subjected to four successive impacts much greater than the regulation drop of 9 m or the drop on to a bar with a diameter of 150 mm.

2.1.1 Experimental conditions

The damaged package was suspended vertically 80 cm above the upper surface of paraffin (kerosene) which was held in a 2 x 4 m tank. The package was positioned to allow any melted lead to pour through the two openings in the casing. The package was fitted with 21 thermocouples and a further two thermocouples were placed 3 cm below the bottom in order to measure the temperature of the flames.

The flask was subjected to fire for 30 minutes. The wind speed varied from 2.5 to 3 m/s. As a result of the wind conditions, the temperature varied and uniform heating of the flasks was not obtained.

2.1.2 Analysis of results

The data indicated that the fire was properly maintained for 30 minutes but that external conditions caused the temperature to vary widely. Two separate periods were clearly distinguished, however:

- 0 to 16 minutes, average temperature 650°C;
- 16 to 30 minutes, average temperature 800°C.

Temperatures at the internal compartment did not exceed 100°C (the temperature of a steam atmosphere). The plug for the internal compartment reached a temperature of 300°C. The container steel in the fire zone reached 650°C, while outside the fire the steel temperature reached 520°C.

Many of the thermocouples showed that the temperature levelled off at 100°C for 17 minutes. This proved that the plaster in the package played its part and that for 17 minutes the energy transmitted served only to evaporate the water in the plaster. A weight loss of 4 kg was observed after the test. The lead temperature reached 200°C after 36 minutes. No melting of lead was observed which is borne out by the X-ray photographs.

2.1.3 Summary of Findings

This fire test on a badly damaged package left its radiation shield virtually untouched: the lead did not melt and the plaster played a decisive role in providing protection.

2.2 IL46 Analysis

In order to evaluate the degree of safety afforded in a fire of long duration by packages designed for the transport of radioactive materials, different thermal transient calculations were applied to a package which is in current use, the IL46 type for transporting spent fuel elements.

Attempts were made during the study to estimate the safety margin. The following results were obtained:

(a) Following a regulation fire (800°C for 30 minutes), the maximum temperature reached by the lead was 160°C and remained far below melting point.

(b) The IL46 package will withstand a 800°C fire for 200 minutes without deteriorating; this time is reduced to 115 minutes for a 1200°C fire.

(c) If the plaster is dried (following an earlier fire, for example), the fire resistance is reduced but still remains in compliance with the regulations.

(d) Wet plaster protects the lead for 200 minutes.

(e) Dry plaster protects the lead for 166 minutes.

2.3 TN-12 Thermal Analysis

The TN-12 [1] is a 100 tonne, steel-walled flask used to carry 12 PWR irradiated fuel elements. The container seals would be the most likely component to fail in a fire. The seals are made of Viton, as per ASTM Code D.2000, whose integrity is not guaranteed beyond a temperature of 316°C. The maximum seal temperature during shipping under normal conditions is deemed to be 155°C for a maximum load carried of 120 kW. Although this calculation was based on a load carried of 120 kW, the load carried rarely exceeds 85 kW.

2.3.1 Calculation method

The DELPHINE Computer Code [2] was used. This is a two-dimensional code which can only be used for analysis of plane or axisymmetric geometries. Hence, a preliminary calculation was made to transform real geometry into thermally equivalent axisymmetric geometry. Only the top section of the container was evaluated by imposing a null flow over the median plane. The container's geometry is not exactly symmetrical with respect to this plane, but the heat sources are. Furthermore, the geometric dissymmetries are far from this plane, hence they only intervene in the edge effect. The assumption of null flow on this plane is hence justified. The top section was chosen given that under constant conditions, it provides the least margin of safety at the seals. Afterwards, the results were extrapolated for the bottom section.

Calculations were performed using a network containing 3031 nodes and 2731 elements. It was assumed that the radiating environment had an emissivity of 1.0 and that the flask surface level an absorptivity of 0.83. Convection in the fire was also considered.

2.3.2 Fire model

The fire was modelled, based on the environmental conditions as follows.

Time	Temperature
0	38°C
3 min.	840°C
30 min.	840°C
Beyond 30 min. to end of exposure	800°C
Beyond end of exposure	38°C

Total fire durations of 30, 60 and 90 minutes were calculated. The calculation was run up to the time where the seal temperature reached a maximum, this maximum was always reached during the cooling down period.

2.3.4 Summary of findings

In the cases studied the seal did not reach its design limits of 316°C. The results of the calculations are summarized as follows:

Exposure Time (minutes)	Maximum Seal Temperature (°C)	Time after extinction when seal begins to cool (hours)
30	183	10
60	199	11
90	213.5	12.5

Thus, for each additional 30 minutes of fire duration, the seal increases in temperature by about 15°C.

Due to the location of the seal evaluated, it is only affected by the fire via the axial flow through the balsa wood. The results show that the characteristic times are shorter than for the top seal (around 6 hours after the extinction of the fire). Extension of the length of the fire is slightly more significant: 17°C for 30 minutes.

It is difficult to draw an accurate picture of the duration of a fire for which a seal would reach the maximum temperature of 316°C during cool down. In the case of long duration fire account must be taken of the radial flow through the balsa which increases during a long fire. This flow corresponds to a transfer of heat accumulated in the balsa to the steel.

The isothermic developments showed that the temperature of the seal will increase during the extended fire period by around 15°C every 30 minutes. The effect of the heat flow in the balsa was estimated at this value. Considering that the flow in steel does not come in on this approach, its effect remains that observed up to now: 15°C per 30 minutes. The overall effect is 30°C per 30 minutes. A conservative estimate gives a 3-hour fire as necessary to reach the seal temperature limit at which integrity is lost. Conversely, if the additional heat from the balsa is ignored, the temperature increase remains at 15°C per 30 minutes. The maximum fire duration then becomes 5 hours.

Thus, the true limit is thus between a 3 to 5 hour fire.

2.4 FS47 Analyses

The French container FS47, Figure 1, is a Type B fissile package used for carrying plutonium oxide (PuO_2). The FS47 container is approximately 2 metres high with a diameter of 0.75 metre, and the mass fully loaded is around 1500 kg. The thermal behaviour of an FS47 container during and after fires of varied duration (30, 60, 90 minutes) and temperatures ranging from 800°C to 1000°C was assessed using the CASTEM system DELFINE code [2].

2.4.1 Calculation method

Development of the container model was carried out using reference measurements made during the thermal test for certification and approval. Simplifications related to the use of the DELFINE code were necessary because it is an axisymmetric model and obstacles were presented by the association of materials whose diffusivity is very different during a sharp transition. The zoning in the model consists of 2626 nodes, 2349 rectangular elements and 81 shell elements. The modelling was similar to that used for the TN-12 flask; however, as noted above, the model was adjusted using test data. An example of the comparison between analyses and experiment is shown in Table 1. It was assumed that the total power given off is 177 W and is evenly distributed through the PuO_2 power.

Three fires were analyzed at 800°C: 30 minutes, 60 minutes and 90 minutes and one 90-minute fire at 1000°C. Ambient temperatures of 38°C were assumed before and after exposure.

2.4.2 Summary of findings

The calculated maximum temperatures reached for the different assumed fire conditions are summarized in Table 2. The increase in the temperature of the fire has only a slight effect on the temperatures inside the container. Conversely, the duration of the fire has a significant influence on the temperature (e.g., on the inner shell 35°C per additional half-hour). The seal temperature under worst case conditions (90 minutes at 1000°C) does not reach 170°C and hence is far from the allowable limit of 315°C.

3.0 EVALUATION OF SAFETY MARGIN OF UF_6 PACKAGE

Uranium hexafluoride (UF_6) is classed under the heading of radioactive matter in the International Transport Regulations. Nevertheless, the UN and the WIO classify UF_6 under class 7 and add a secondary hazard: the UN states corrosion as the secondary hazard (class 8); formerly the WIO stated toxicity as the secondary hazard (class 6.1) but as of January 1985, the WIO aligned itself on the UN's classification giving corrosion as the secondary hazard (class 8).

Natural or slightly enriched (less than 1%) UF_6 is a material which presents a very slight radiation hazard (given the low specific activity of uranium) and also presents a chemical hazard. UF_6 breaks down into uranium oxifluoride (UO_2F_2) and hydrofluric acid (HF) in humid air or water. HF gas is highly toxic. The chemical hazard is increased if the UF_6 is in a liquid or gaseous state. UF_6 enriched 1% or less is transported in its solid state in an "industrial" container "US48Y" (DV08 in France). UF_6 enriched by more than 1% is transported in a Type B container ("US30B fitted with a 21PFT shell; DV05 and FS50 in France respectively).

Under radioactive transport regulations, industrial containers must meet certain requirements; however, they are not required to be tested in a 30-minute fire at 800°C, as is the case for the 30B containers which are Type B containers. The DV08 steel container (Figure 2) is a horizontal axis cylinder, the outer diameter is 121.9 cm (48 inches), the length is 381 cm, and the wall thickness is 15.9 mm. It has a guard rim at each end, a fill valve at one end and a plug at the other for draining upon cleaning or other operations. Its maximum capacity is 12,501 kg and total mass loaded is 14,860 kg. The valve body is made of cupro-aluminium (3.5% Al and 1% Si) and the valve is made of inconel. Service pressure is 16 bars, test pressure 28 bars.

TABLE 1

COMPARISON OF MEASUREMENTS WITH CONTINUOUS AND TRANSIENT STATE CALCULATIONS
(Temperatures in °C)

LOCATION	MEASURED	CONTINUOUS STATE CALCULATION	TRANSIENT STATE CALCULATION AFTER 18 h
Ambient	8	8	8
Outer shell	11	11	10.8
Copper shell:			
top	11	13.7	13.5
middle	13	15	14.9
bottom	13	13.8	13.9
Inner shell:			
top	13	14.4	14.2
middle	17	17.9	17.8
bottom	15	15.8	15.6
Container:			
top	14	16.4	16.1
middle	43	43.8	43.7
bottom	36	35.7	35.4
Top casing	70	68.7	68.5
Bottom casing	93	91	90.9
Top cage	100	103	102
Bottom cage	145	140.5	140.5
Top box			
top	116	122	122
bottom	171	169	169
Other boxes		183	183

TABLE 2

TEMPERATURES REACHED DURING THE FIRE FOR THE FS47

Fire Temperature	800°C						1000°C	
Fire Duration	30 mn		1 h 00 mn		1 h 30 mn		1 h 30 mn	
Location	Max. °C	Time	Max. °C	Time	Max. °C	Time	Max. °C	Time
Outer shell	784	30 mn	786	1 h 00 mn	786	1 h 30 mn	990	1 h 30 mn
Copper shell	103	30 mn	148	1 h 00 mn	188	1 h 30 mn	225	2 h 30 mn
Inner shell	96	1 h 30 mn	134	2 h 00 mn	170	2 h 00 mn	203	2 h 00 mn
Container	122	4 h 00 mn	151	4 h 00 mn	181	4 h 00 mn	206	3 h 45 mn
Casing	131	5 h 00 mn	156	5 h 30 mn	181	5 h 00 mn	225	5 h 00 mn
Cage	162	6 h 00 mn	184	5 h 30 mn	206	5 h 30 mn	254	5 h 45 mn
PuO_2 box	237	6 h 00 mn	249	5 h 30 mn	268	5 h 30 mn	284	5 h 45 mn
Plug	83	3 h 30 mn	113	3 h 30 mn	143	3 h 30 mn	168	3 h 30 mn

3.1 Preliminary Assessment

Calculations and tests regarding the behaviour of a full container in a fire have been done some years ago. The results are given.

(a) Local heating of container - Under this assumption, a break could occur upon liquification of the UF_6 in a localized region due to the increase in volume (around 30%). An experiment run on a container filled with UF_6 has shown that the mass of UF_6 was sufficiently cracked to enable the diffusion of the liquid and vapour phases in the solid. Calculations show that no abnormal local stress should occur at the wall.

(b) Heating of the entire container - Calculations were run based on the results of American experiments on small containers 24, 25, 111 and 113 kg of UF_6, to estimate container 48Y strength. The calculated times before rupture, with the introduction of the most conservative parameters are 61 minutes at 800°C and 47 minutes at 900°C.

(c) Valve behaviour - Tests simulating a fire of 800°C were also run in a specifically outfitted oven, on models consisting of the valve mounted on the corresponding container section. During these tests at an internal pressure of 5 bars, the start of a leak occurred when the valve reached 205°C due to the fusion of the tin-lead alloy deposit used to seal the valve-container interface. Leakage was slight for valve temperature of less than 400°C [3].

After the Mont Louis accident in the North Sea it was decided to re-evaluate the behaviour of this kind of packaging under different conditions of fire environment to take into account fire on board a ship (200°C to 900°C).

3.2 Calculation Method

Assessment of the pressure-temperature phases of UF_6 shows that the triple point is at 64°C and 1134 mm of Hg. Thus, even at low temperatures the three phases will occur during container heat-up. The liquid phase will show up at the perimeter, thereby changing the nature and intensity of the heat exchange between the wall and the UF_6. Moreover, the denser solid core will sink to the bottom, causing significant dissymmetry of exchanges and compensatory movements.

The purpose of the modelling was not to create a detailed model of the complex phenomena but rather to determine a simple model enabling an accurate determination of internal pressure over time. The approach consisted of formulating two simplified assumptions:

- the temperature inside the container is uniform: this assumption is not a conservative one but is allowable since the apparent thermal conductivity of UF_6 is preponderately a sublimation exchange mode, gas flow and resolidification;

- the heat exchange between the container wall and the UF_6 in contact with the wall is perfect; this assumption is conservative since UF_6 in contact with the wall will liquify and exchange will occur via convection.

The container was assumed to initially be at a uniform temperature of 20°C. The container exchanges heat with the hot environment via natural convection and radiation. At each temperature step, a check is made as to whether there has been a slight change in the exchange coefficient over the time interval. The ambient temperature was varied from 200°C to 900°C.

The maximum fill percentage is based on a minimum volume of 4.04 m^3 for DV08 containers which corresponds to a mass of 12,500 kg. This value was used to run a parametric analysis on ambient temperature. Also a parametric analysis on the percentage of capacity for ambient temperatures of 200°C, 400°C and 800°C was made with the following results:

	(theoretical)
14,500 kg	70.5%
12,500 kg	60.5%
10,500 kg	51.1%
8,500 kg	41.3%
6,500 kg	31.6%
4,500 kg	21.9%

3.3 Summary of findings

Figure 3 gives the graph of container internal pressure with respect to internal temperature. For a theoretical high percentage of fill (14,500 kg), internal pressure is due solely to compression of the entrapped air due to the expansion of the volume of UF_6; saturation pressure at 70°C is 1.5 bars whilst internal pressure is 32 bars.

The lower the percentage of fill, the less significant this phenomenon is. For 12,500 kg of UF_6, for nominal fill capacity air compression and UF_6 saturation pressure act equivalently in the increase of internal pressure, which is 28 bars at an ambient temperature of 140°C. From 10,500 kg downwards the curve approaches the saturation curve, and saturation pressure becomes the sole factor and the percentage of fill is no longer a factor. These results are paradoxical since in certain cases, the pressure limit is reached faster in less filled containers. This can be explained by the fact that if the temperature to reach is higher, thermal inertia is lower. Thus, the fill percentage should not be lower than 10,500 kg.

This analysis has shown the influence of the two major parameters involved in the thermal behaviour of a DV08 container filled with UF_6; that is, fire temperature and fill percentage. Under normal loading conditions (12,500 kg in a 4.04 m^3 container), the container will withstand a 200°C fire for around 20 hours, under the same conditions it will only withstand an 800°C fire for 30 minutes. If the percentage of fill is lower, the time required for the pressure limit to be reached is not necessarily extended since the thermal inertias are decreased.

4.0 EVALUATION OF SAFETY MARGIN OF TYPE A (ELUMATIC III) PACKAGE

The ELUMATIC III is an automatic and highly protected system for easily obtaining a sterile apyrogenic solution of ^{99m}Tc in the form of sodium pertechnetate. This solution is eluted from an aluminium chromatographic column to which fission ^{99}Mo (T = 66h) has been fixed. This material is a radioactive ascendant of ^{99m}Tc (T = 6.02 h).

The ELUMATIC III is delivered in a sealed steel drum, which is a Type A container. The activity is written on the ELUMATIC III label stating the ^{99m}Tc available at the date of calibration, which varies from 50 to 500 mCi (1.85 to 18.5 G Bq) according to order.

Regulations obviously submit Type A containers to stringent resistance requirements; however, they are less severe than for Type B containers, especially regarding the 30-minute 800°C fire test which is not required for Type A packagings.

4.1 Container description

The system, Figure 4, consists of:

- a flexible plastic pouch containing the elulate solution (1). It is connected by a stainless steel needle (2) to the top of the chromatographic column.
- a glass chromatographic column (3) fitted with a filter (4) at its base for removing the alumina contained by the column. It is closed at both ends by plugs held in place by aluminium capsules (5). The column contains alumina (6) to absorb the molybdate ions and is inert with respect to the pertechnetate ions.
- an outlet needle (7), one end of which is connected to the base of the column. The other end (8) can be fitted with a vacuum flask or a flask containing a bacteriostatic agent for maintaining sterility between two elutions. The column and the needles are protected by lead shielding (9) in the form of a conic cylinder, minimum thickness is 52 mm.

The lead shield-plastic pouch assembly is placed in a parallelepipedic sealed moulded nylon packing (23 x 23 x 14 cm) (10). On top of the packing is the elution stage closed by a cylindric container B enclosing the bacteriostatic liquid flask (11) in position on the outlet needle. This flask contains an aqueous solution. Container sealing is ensured by an O-ring (12). Next to the elution stage is a pan for the elution flasks. It contains a safety valve (0) which is closed during transport.

The ELUMATIC III is delivered in a sealed steel container.

4.2 Analysis of the thermal behaviour of sodium molybdate

Thermogravimetric and differential thermal analyses enable observation of the phenomena referred to in this field of study: phase changes in solid state, melting point: 690°C, a decomposition point is not observed up to 800°C; there is no product mass loss during the duration of the phase (2 h) at this temperature.

The sodium molybdate deposited on a porous alumina column was analyzed under the same conditions (800°C for 2 h); a continuous mass loss was observed with respect to temperature, which would indicate water loss even at high temperatures.

4.3 ELUMATIC III fire resistance

An ELUMATIC container was fitted with 11 thermocouples and exposed to a pool fire for 30 minutes. An additional thermocouple was placed between the heat source and the container to obtain the flame temperature. The mean flame temperature was around 750°C.

After about 20 minutes, a sharp drop in temperature occurred, which corresponds to the time when the elulate solution spreads into the container when the plastic pouch (1) burns; the temperature then rose sharply. This phenomenon was observed over several temperature reccrdings.

At the end of 1 minute the cover opened and the flames penetrated the container causing the polyethylene foam and nylon packing to catch fire. After 15 minutes of fire the lead began to melt at 326°C (the melting point of pure lead is 324°C) and it flowed out of the container for 5 to 10 minutes. Again the temperature dropped sharply at the end of 20 minutes and then later increased.

At the end of 24 minutes the glass column (rebaked Pyrex) containing alumina on which the fission ^{99}Mo is fixed reached a temperature of 675°C. This temperature remained nearly stable up to the end of the test.

After complete cooling, the container was examined. The nylon packing and polyethylene foam were completely consumed. The 13.3 kg of lead had completely flowed out; only the Pyrex column remained, which was open due to slight deformation. The plugs held in place by the aluminium capsules had burned. The alumina seemed to be intact. The sodium molybdate remained fixed to the alumina.

4.4 Summary of findings

The ELUMATIC III lead container no longer ensures biological protection at the end of 25 minutes in an 800°C fire. Given the melting point of sodium molybdate is 690°C and that the temperature reached at the chromatographic column is 675°C after 24 minutes of fire, it would be reasonable to conclude that the sodium molybdate has remained fixed to the alumina.

5.0 CONCLUSIONS

The studies have demonstrated that not only Type B packages, but those having less stringent design criteria, are capable of inherently withstanding accidental fire exposure well in excess of their design values with little radiological consequences. One of the Type B packages was exposed to extra-severe mechanical tests prior to exposure to the regulatory thermal test without severe damage to its radiation shield. Three other Type B packages were assessed for longer duration and for hotter environments than required by regulations. For 800°C exposures, safety margins (that is, times to failure above the required 30 minute exposure) ranging from 130 to 270 minutes were identified. Exposure to higher temperatures reduces those margins, but the survival exposure times were in excess of 30 minutes (e.g., 85 minutes for the IL46 package when exposed at 1200°C).

The Industrial Package (the UF_6 package) was assessed to be able to withstand a 200°C exposure for approximately 20 hours and exposure at 800°C for 30 minutes.

The Type A package assessed, although severely damaged in a 30 minute pool fire test, still retained its contents.

6.0 REFERENCES

[1] Drawing 9452-167 TRANSNUCLEAIRE

[2] System: CEA SEMT - DELPHINE
User's Manual - CEA/SMTS/78.25

[3] Etude du comportement dans un fer à 800°C de la vanne d'un conteneur d'UF_6 - rapport CEA/STT/STML/83.23 du 29/04/1985.

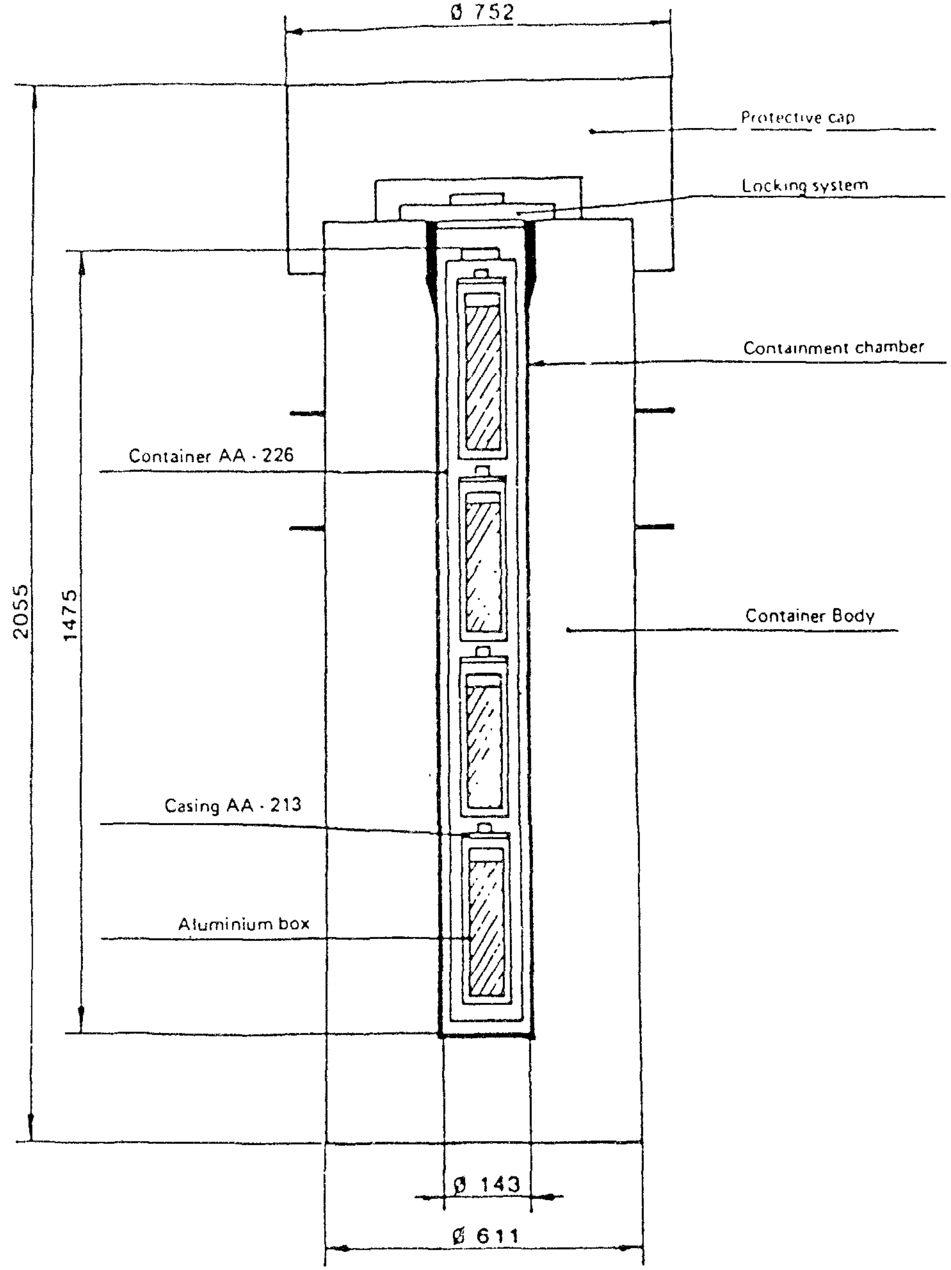

Figure 1
The FS47 Container

MASS 2 359 kg

USEFUL CAPACITY 4 040 liters for 12 500 kg of UF6

Maximum enrichment allowed for during
transport . less than 1 %

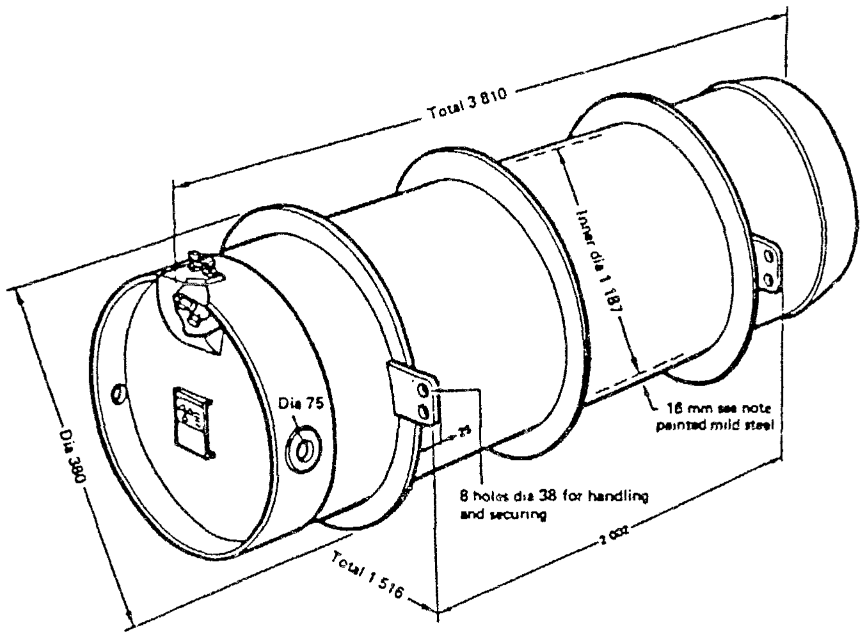

Figure 2
DV08 (US48Y) Container
for UF_6

Note the 5 DV-08 numbered 026, 027, 028, 029, 030 have a wall thickness of 8 mm – masse 1 450 kg
and are used for storage

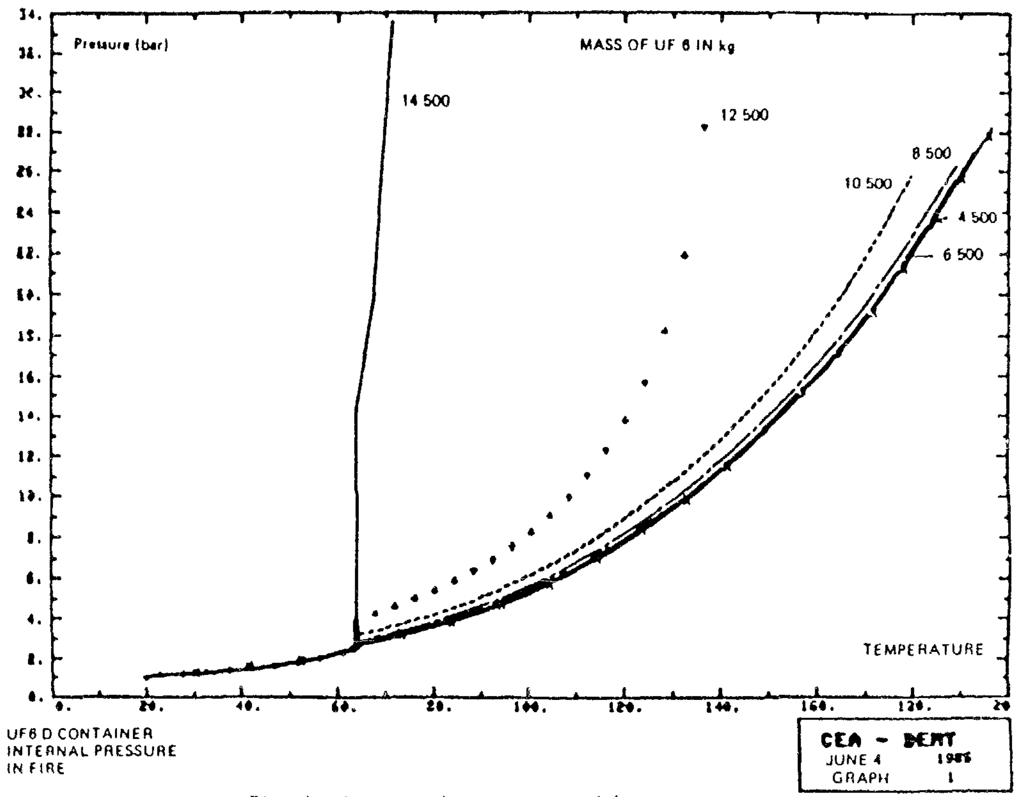

Fig. 3 - UF_6 container pressure with respect to internal temperature

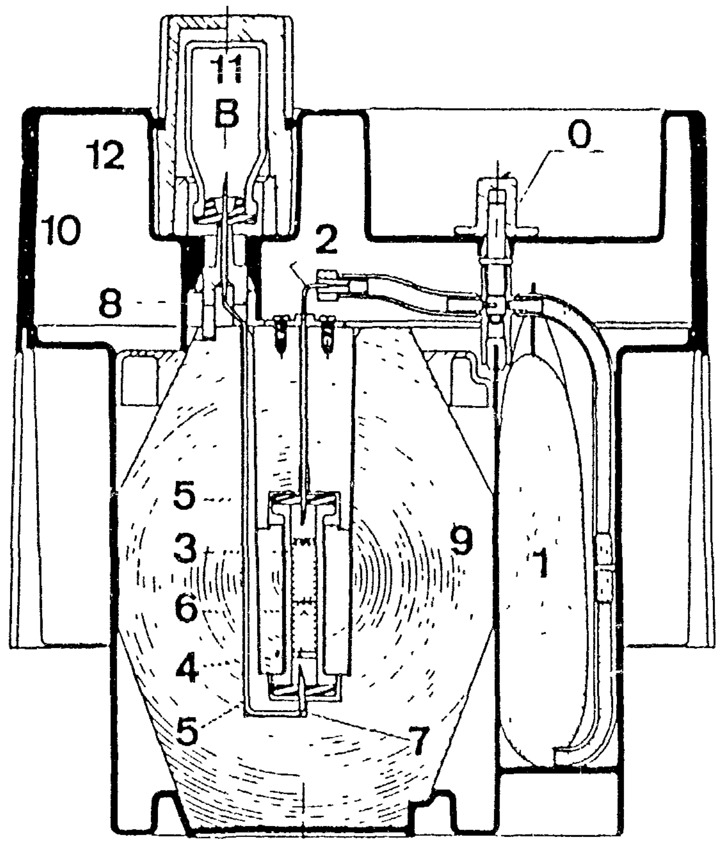

Fig. 4 The FLUMATIC III system

MODEL INVESTIGATIONS OF THE HEAT TRANSFER IN DRY SPENT FUEL CASKS*

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1.0 INTRODUCTION

The transportation of spent fuel assemblies (SFAs) has to meet high safety requirements laid down in the GDR in the Decree on the Transport of Radioactive Materials [1] in accordance with the IAEA Regulations for the Safe Transport of Radioactive Materials [2]. Under this legislation, particular attention has to be focussed on the safe dissipation to the environment of the heat generated by the SFAs without impermissible temperatures developing at the shipping cask or the SFAs. Compliance with this requirement has to be demonstrated by determining the temperature distribution in the cask including the maximum surface temperatures of spent fuel elements (SFEs) and SFAs as main safety parameters [3].

The cask usually contains several SFAs, whose afterheat is transferred to the cask wall by a gaseous coolant. In general, cask wall temperatures can be calculated or measured with a sufficient degree of accuracy [3] but maximum SFA and SFE surface temperatures cannot be measured directly in the cask. So far, it has only been possible to measure the surface temperature of a selected SFA in its original cask, using a special experimental programme [4].

For a system of several SFAs in the cask, the calculation of the maximum SFA and SFE surface temperatures is very difficult because of the relatively complicated arrangement of heat sources and because the heat transfer processes occur simultaneously due to free convection, conduction and radiation. For such conditions no reliable calculation methods are currently available to determine these parameters with a high degree of accuracy. Therefore, model experiments using electrically heated rods were performed to investigate the heat transfer processes between SFAs and cask wall as well as between SFEs in the SFA and to determine maximum SFA and SFE surface temperatures in dependence on various parameters. A further aim was to check heat transfer calculation methods which can be applied to the original cask after verification by model experiments.

The present report describes the investigations made and the results obtained under the IAEA Research Agreement No. 2954/CF by May 31, 1985. The results of the first year period of the Research Agreement were published in [5] and discussed at the IAEA Research Coordination Meeting in Rome in 1983. The present report is to outline the research work continued in the second and third years of the Research Agreement and describes the experimental programme, the measuring results obtained and their comparison with calculation methods.

* Research Agreement No. 2954/R1/CF.

2.0 MODEL DEVELOPMENT, OBJECTIVES AND RESEARCH PROGRAMME

The cylindrically shaped original cask has a 350 mm steel shielding and contains 30 spent WWER-type fuel assemblies hexagonally arranged in a basket [4,6]. The afterheat of the SFAs has to be dissipated to the environment through the cask wall by several heat transfer processes described in detail in [5].

Heat transfer in the cask (internal heat transfer) is considered to be a radial heat exchange process in which the afterheat of the SFEs is transferred to the SFA wall and then to the inner wall of the cask. For the purpose of model experiments this process can be regarded simply as taking place between an arrangement of heat-dissipating rods and their surrounding wall. Therefore, the SFAs and SFEs can be simulated by electrically heated rods and model experiments performed in 2 stages:

1. Investigation of the heat transfer from the SFAs to the inner wall of the cask using a model arrangement of 30 heating rods,

2. Investigation of the heat transfer from the SFEs to the SFA surface with a SFA scale model using 90 heating rods.

The investigations made and the results obtained in the first stage are represented in [5]. They allowed the maximum SFA surface temperature in the shipping cask to be determined and led to a better qualitative insight into the processes of heat exchange, to a more precise evaluation of internal temperatures measured in the original cask and to a quantitative description of radiant heat transfer between the SFAs and the cask wall.

In the present second stage, heat exchange processes between the SFEs inside a SFA are studied by using a SFA model and determining the maximum SFE surface temperature in dependence on heat rates and wall temperatures of the SFA. For this purpose, the temperature distribution of the SFA model is measured under different environmental conditions in accordance with the following experimental programme:

1. Measurement in free air environment in the vertical and horizontal position of the SFA model,

2. Measurement in the model cask under vacuum for the determination of radiant heat transfer and comparison with calculations,

3. Measurement in the model cask under normal pressure and overpressure using different gaseous coolants for determining total heat transfer and comparison with calculations.

In the following experimental setup and results are described. Investigations of Point 1 and 2 have been completed. For Point 3 the results obtained by May 31, 1985, are presented, since these investigations are to be completed by March 31, 1986, in accordance with Research Agreement No. 2954/CF.

3.0 EXPERIMENTAL SETUP

3.1 Model of spent fuel assembly (SFA model)

Fig. 1 shows the SFA model. It represents a model highly reduced in length of a SFA of the WWER-type pressurized water reactor having 90 fuel elements. The cross section of the SFA is modelled nearly on an original scale (scaling factor 1.3). The fuel elements are simulated by electrically heated rods. They are hexagonally arranged according to Fig. 2 and they are positioned by 3 spacer grids at the upper and lower end and in the middle of the SFA model. The temperature distribution of the SFA model is determined by means of iron-constantan thermocouples at selected measuring points as can be seen from Fig. 2. The radial temperature profile is measured at medium height of the heating rods and the vertical temperature distribution at the innermost heating rod No. 5 and at the SFA wall. The coolant temperature is measured by two thermocouples at the upper and lower end of the SFA model. By means of these selected measuring points representative temperature distributions of the SFA model are determined, which seem to be adequate for the comparison with calculation methods.

3.2 Experimental equipment

The experimental equipment is shown schematically in Fig. 3. In its main components it corresponds to the equipment already described in [5]. It was designed for measurements under vacuum and under overpressure in the model cask. The experimental equipment mainly consists of:

(a) model cask (internal diameter 180 mm, height 600 mm, 2 mm wall of stainless steel) pivoted in a frame in such a way that measurements can be made in vertical and horizontal positions. It has been designed for an overpressure up to 0.7 MPa;

(b) the SFA model described under point 3.1 which is contained in the model cask;

(c) the vacuum system consisting of a pump for low vacuum, sorption trap, various valves and diffusion pump for high vacuum. Thus, in the model cask a vacuum of up to 7 mPa ($5 \cdot 10^{-5}$ torr) can be attained;

(d) the pressure system consisting of a safety valve, pressure gauge, pressure connection for overpressure measurements in the model cask from 0.1 to 0.7 MPa with various gaseous coolants;

(e) devices for power supply, heat rate measurement and temperature measurement and registration.

Using this experimental equipment, measurements were made under vacuum to separately investigate the radiant heat transfer in the SFA model and, under normal pressure and overpressure, to investigate the total heat transfer including convection and conduction.

4.0 RESULTS

4.1 Temperature distribution of the SFA model in free air environment

In the first test series the temperature distribution of the SFA model was measured in free air environment for various heat rates in a vertical and horizontal SFA position to investigate these heat dissipation processes which are of importance for reloading and handling operations.

Fig. 4 shows the radial temperature distribution of the SFA model measured in a vertical position for different heat rates at medium height of the rods and the SFA wall. For reasons of symmetry, there is the same temperature profile on the right side of Fig. 4 as on the left, which is not represented in the Figure. This radial temperature distribution is characterised by a very small temperature difference between the innermost heating rods No. 5 and 4 and a strong temperature decrease from the inner rods to the SFA wall. The measurement of the vertical temperature distribution at the innermost heating rod No. 5 showed that, due to convective heat exchange, the maximum surface temperature does not occur at medium height of the rods but in the upper third of the heated zone of the rod. Relative to the heated length of the rod l_H = 410 mm, the temperature maximum lies roughly at $l_H/4$ above the middle of the heating rod. This maximum temperature is maximally 7% higher than that measured at the medium height of the rod. It represents the desired maximum heating rod surface temperature in the SFA model and is shown in Fig. 6 in dependence on the heat rate of the SFA. Fig. 5 shows the temperature distribution in the horizontal position of the SFA model. In contrast to the vertical position, there is no symmetric temperature distribution. The temperature maximum is shifted from the central axis of the SFA towards the upper third. This shifting is caused by the free convection flow perpendicular to the heating rods in the SFA model. Hence, there are higher temperatures at the upper than at the lower side of the SFA wall. The maximum heater surface temperatures in horizontal position measured at the middle of the rods are also given in Fig. 6. In comparison with the measuring results in vertical position it can be seen that in both cases there is about the same maximum surface temperature of heating rods in the SFA model. Here, it should be taken into account that the vertical flow through the SFA is reduced due to the design of the spacer grids in the SFA model. Therefore, for SFAs with a favourable vertical flow of gaseous coolant, lower maximum temperatures can be expected for the vertical position than for the horizontal one. On the whole, from the viewpoint of thermal engineering, the horizontal and vertical position seems to be nearly equivalent for the safe handling of SFAs in free air environment.

4.2 Temperature distribution of the SFA model under vacuum in the model cask

The radiant heat exchange in the SFA model has to be considered as a dominant heat dissipation process for the resulting temperature distribution. For its separate investigation, the SFA model was inserted into the model cask, the model cask was evacuated and measurements were made in the pressure range from 700 mPa ($5 \cdot 10^{-3}$ torr) to 13 mPa (10^{-4} torr).

The emissivity of the heating rod and the SFA surface required for calculation was measured separately. The emissivity of the heating rods of stainless steel ε_H was measured calorimetrically according to [8] for a single centrally arranged heating rod in the model cask at a pressure of 8 mPa ($6 \cdot 10^{-5}$ torr). It was found to be $\varepsilon_H = 0.39 \pm 0.03$. The emissivity of the SFA wall of stainless steel was measured with an optical method and yielded $\varepsilon_{SFA} = 0.35 \pm 0.03$.

4.2.1. Measuring results and comparison with calculations of radiant heat transfer

The radial temperature distribution of the SFA model measured under vacuum for different heat rates can be seen in Fig. 7. These experimental results are compared with calculations, using the computer programme for radiant heat transfer described in [5] and [7]. This computer programme [7] calculates the maximum surface temperature of each heating rod in the SFA at a given heat rate per rod and SFA wall temperature. Fig. 7 shows that the calculated temperatures are maximally 8% higher than those measured in the low temperature range. The agreement between calculation and experiment becomes systematically better with increasing surface temperatures of heating rods. This indicates that at low temperatures the heat transfer by thermal conduction of the residual gas in the SFA model cannot be neglected and contributes to a noticeable reduction of the maximum surface temperature of heating rods as compared with pure radiant heat transfer. Taking into account errors due to theoretical simplications, measuring errors (maximum relative error for heater surface temperature from ± 3% to ± 7% depending on temperature) and the fraction of radiant heat transfer in the total heat transfer, the obtained agreement between calculation and experiment can be considered sufficient for the aims of safety assessment and technical design.

Using the emissivities of the original SFA, the computer programme can be directly applied to it for determining the SFE temperatures for pure radiant heat transfer in dependence on heat rate and wall temperature of the original SFA.

4.2.2. Simplified calculation of radiant heat transfer and generalization of the results

For safety considerations and technical applications it is very often desirable and adequate to determine only the maximum SFE surface temperature in the SFA (rod No. 5) in a quick and simple manner and it is not necessary to calculate the temperature distribution in detail. For these purposes a simplified analytical representation for the radiant heat transfer between the innermost heating rod (No. 5) and the SFA wall is required. Such a representation can be derived on the basis of the Stefan-Boltzmann law as follows:

$$\dot{Q}_H = C_s \cdot \mathcal{F}_M \cdot F_H \cdot [(T_{max}/100)^4 - (T_{SFA}/100)^4] \qquad (1)$$

where

$\dot{Q}_H$ = heat rate of a heating rod ($\dot{Q}/90$),

C_s = radiation constant of the black body = $5.78 \cdot 10^{-4} \dfrac{W}{cm^2\ K^4}$

$\mathcal{F}_M$ = radiant configuration factor for SFA model,

F_H = heat transfer area of the heating rod,

T_{max} = maximum surface temperature of heating rods in the SFA model (innermost heater No. 5) in K,

T_{SFA} = SFA wall temperature in K.

The radiant configuration factor has to be determined. It was calculated from (1) by means of T_{max}, T_{SFA} and $\dot{Q}_H$ measured in experiment. Thus, by way of model experiments the theoretical value of $\mathcal{F}_M$ = 0.036 derived from the calculations by the computer programme could be confirmed up to an accuracy of 6%.

Using this value, the following simplified representations for the radiant heat exchange in the SFA model can be derived from (1):

- for the transferable heat flow density $\dot{q}_H$ of the innermost heating rod (No. 5) having the temperature T_{max}:

$$\dot{q}_H = \dot{Q}_H/F_H = 2.08 \cdot 10^{-5} \frac{W}{cm^2\ K^4} [(T_{max}/100)^4 - (T_{SFA}/100)^4] \qquad (2)$$

- or related to the transferable heat rate of the entire SFA model containing 90 heating rods:

$$\dot{Q} = 90\ \dot{Q}_H = 0.235 \frac{W}{K^4} [(T_{max}/100)^4 - (T_{SFA}/100)^4] \qquad (3)$$

From (3) also the desired maximum surface temperature of heating rods in the SFA model can be calculated in dependence on SFA wall temperature and heat rate:

$$T_{max} = 100 \sqrt[4]{\frac{\dot{Q}}{0.235} + \left(\frac{T_{SFA}}{100}\right)^4} - 273 \qquad (4)$$

where

T_{max} = maximum temperature of heating rod surfaces in the SFA model in °C,
T_{SFA} = SFA wall temperature in K,
$\dot{Q}$ = heat rate of the SFA model.

Fig. 8 shows the surface temperatures of heating rods calculated according to (4) along with the T_{max} measured and the respective SFA wall temperatures T_{SFA} and temperatures of the model cask T_W. Due to the influence of residual gas thermal conduction, the calculated temperatures lie maximally 8% in the lower range and only 2% in the upper range above those measured. This result corresponds to the conditions described under point 4.2.1.

Thus, a simple representation for calculating radiant heat transfer in the SFA model has been found. However, it cannot be directly applied to the original SFA, since other emissivities have to be considered for this. Equations (2) to (4) are valid only for the emissivities of the model experiment ($\varepsilon_H = 0.39$, $\varepsilon_{SFA} = 0.35$) and therefore they have to be generalized for any ε_H and ε_{SFA} values. The dependence of ε_H and ε_{SFA} has to be taken into account by the radiant configuration factor $\mathcal{F}(\varepsilon_H, \varepsilon_{SFA})$ and can be represented according to (1) as follows:

$$\dot{Q}_H = C_s \cdot \mathcal{F}(\varepsilon_H, \varepsilon_{SFA}) \cdot F_H \left[(T_{max}/100)^4 - (T_{SFA}/100)^4\right] \quad (5)$$

This radiant configuration factor $\mathcal{F}(\varepsilon_H, \varepsilon_{SFA})$ in (5) was determined by using the computer programme [7]. The resulting function $\mathcal{F}(\varepsilon_H, \varepsilon_{SFA})$ in dependence on the emissivities is shown in Fig. 9. It can be seen that the emissivity of the SFA wall has a greater influence on the radiant figuration factor than the emissivity of the SFEs. Thus, for increasing the heat removal by thermal radiation above all a SFA wall with a high emissivity is effective. Equation (5) and Fig. 9 can be used to calculate for any values of ε_H, ε_{SFA} and T_{SFA} either the maximum surface temperature of heating rods T_{max} at a definite heat rate $\dot{Q}_H$ or the heat flow dissipated by radiation at a definite temperature T_{max}. Compared with the model experiments, this simple method makes it possible to calculate either the radiant heat flow to an accuracy of about 10% or the maximum temperature of heating rod surfaces to an accuracy of about 5%. Thus, a simple general expression for calculating the radiant heat transfer in the SFA model has been found, which can also be applied directly to the original SFA.

4.3. Temperature distribution of the SFA model under normal pressure and overpressure in the model cask

After the radiant heat exchange in the SFA model can be determined according to point 4.2, the next step is to investigate the heat dissipation by free convection and thermal conduction in the SFA model. The convective heat transfer is studied by increasing the internal pressure of the cask and the conductive heat transfer by using different gaseous coolants (air, argon, helium).

In the following the measuring results obtained so far are presented. The temperature distribution of the SFA model was measured in dependence on the heat rate in vertical position of the model cask at an internal cask pressure of 0.1 to 0.7 MPa, using air as gaseous coolant. Fig. 10 shows the radial temperature distribution of the SFA model in the model cask measured under normal pressure (0.1 MPa) for various heat rates. The heater surface temperatures in the model cask are higher than those measured in free air environment (cp. Fig. 4). This temperature increase is caused by the air gap between SFA wall and model cask, which gives rise to an additional thermal resistance for the heat dissipation. The comparison with Fig. 4 also shows that the temperature profile of the SFA model in the model cask is qualitatively similar to that in free air environment and that there is roughly the same temperature difference between innermost heating rod and SFA wall in free air and model cask. This confirms that there is only a small vertical convection flow through the entire SFA and that the radial heat exchange is the dominant heat transfer process in the SFA model.

Fig. 11 shows the temperature distribution of the SFA model measured for two selected heat rates at increasing internal cask pressure of 0.1 (normal pressure), 0.3, 0.5 and 0.7 MPa. It can clearly be seen that, due to pressure increase in the model cask, the heat removal by convection is considerably improved, causing a strong decrease in SFA temperatures. The radial temperature profile changes also qualitatively with increasing pressure. The temperatures shown in Fig. 11 represent the radial temperature profile measured at medium height of the SFA model. However, as already described under point 4.1, they do not correspond to the maximum surface temperatures of the SFA model. These maximum temperatures could be determined from the measurement of the vertical temperature distribution at the heating rod No. 5, the SFA and the model cask wall. With increasing pressure, the temperature maximum is shifted towards the upper ends of the heating rods, exceeding the surface temperature at medium height by 3% to maximally 11%. These measured maximum temperatures of the innermost heating rod, SFA wall and model cask are shown in Figs. 12 to 15 for the various pressures in dependence on the heat rate. These figures also include the air temperatures measured centrally above the SFA model (cp. Fig. 2). It can clearly be seen how the temperature differences between model cask, SFA wall and innermost heating rod become smaller and smaller with increasing pressure and especially how the air temperature approaches more and more the maximum surface temperature of the heating rods in the SFA model. Finally, at an internal cask pressure of 0.7 MPa (Fig. 15), no difference can be found between maximum surface temperature of heating rods and air temperature within experimental error in the lower temperature range and only a slightly lower air temperature in the upper temperature range.

Using the results of point 4.2, the heat fraction transferred by thermal radiation can be calculated for the maximum temperatures measured. It appears that under normal pressure and in dependence on temperature 30 to 45% of the heat rate are dissipated by thermal radiation. By increasing the pressure up to 0.7 MPa this fraction of thermal radiation is reduced to only about 7 to 10%.

The results in Figs. 12 to 15 also demonstrate that the permissible heat rate of the SFA at a certain constant maximum SFE temperature can be considerably increased by increasing the pressure.

The strong pressure dependence of the maximum surface temperature of heating rods, SFA wall temperature and air temperature is shown in Fig. 16 for a selected heat rate of 140 W. The following conclusions can be drawn from this figure:

1. The air temperature approaches the maximum heater surface temperature very quickly with increasing pressure. Already at a pressure of 0.3 MPa it lies less than 10% below the maximum heater surface temperature. Thus, by measuring the air temperature above the SFA at increased pressure, the maximum surface temperature of the heating rods can be determined approximately.

2. With increasing pressure, temperature changes become smaller and smaller. Therefore, a pressure increase beyond 0.7 MPa do not induce effectively any further temperature decrease. For practical application, above all pressure increases in the range from 0.1 to about 0.5 MPa can be used to effectively improve heat dissipation.

5.0 SUMMARY AND CONCLUSIONS

For the investigation of the heat transfer in a spent fuel assembly under transportation conditions, model experiments were performed using an electrically heated SFA model of a pressurized water type reactor. The radial and axial temperature distribution of the SFA model was measured for different heat rates in free air environment in horizontal and vertical position and in the closed model cask in vertical position under vacuum, normal pressure and overpressure, using air as gaseous coolant. These experiments allowed the separate investigation and description of heat transfer processes by thermal radiation, thermal convection and conduction and the determination of the maximum SFE and SFA surface temperature in dependence on various parameters. From the investigations made the following conclusions can be drawn:

- When handling the SFA model in free air environment, roughly the same maximum surface temperatures of heating rods are obtained for the horizontal and vertical position.

- The radiant heat exchange between the fuel elements in the SFA can be calculated with sufficient accuracy by the computer programme [7] or the given simplified analytical representation. Taking into consideration the fraction of heat radiation in total heat transfer, this method allows to approximately calculate the maximum surface temperature of fuel elements in the SFA in dependence on heat rate and SFA wall temperature.

- By increasing the internal cask pressure, the heat transfer in the SFA can be considerably improved and, thus, the maximum SFE temperature considerably reduced. Heat transfer by convection becomes the dominant process. For practical application, above all pressure increases in the range from 0.1 to 0.5 MPa are effective for improving heat dissipation.

- With increasing pressure, the central air outlet temperature of the SFA very quickly approaches the maximum SFE temperature. For pressures above 0.3 MPa, the temperature difference is smaller than 10% so that the maximum SFE temperature can be approximately determined by measuring the air outlet temperature.

In accordance with the Research Agreement, research work will be continued according to schedule and completed by March 31, 1986. For this purpose, at present a computer programme for determining the temperature distribution of the SFA model under normal pressure and overpressure is being developed and the calculated temperatures compared with those measured. The measurements will be supplemented for the gaseous coolants argon and helium and completed in accordance with point 3 of the experimental programme (see point 2).

Altogether the results obtained under Research Agreement No. 2954 provide a better understanding of the mechanism of internal heat transfer in a transport cask and permit a more accurate prediction of fuel element surface temperatures. They contribute to the investigations of the basic heat transfer processes as they have to be considered for the thermal assessment of contents/package behaviour in transportation but also for thermal problems in handling operations and in the dry storage of spent fuel assemblies in a cask or can.

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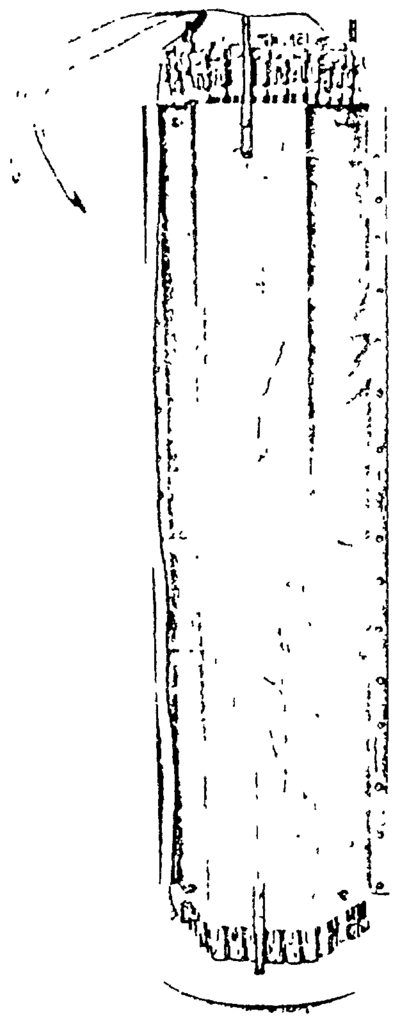
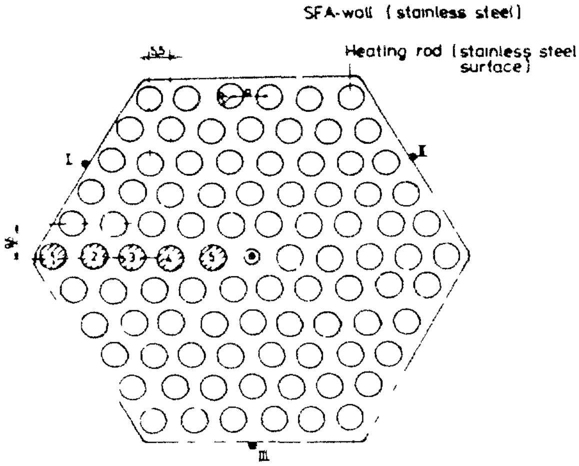


Fig. 1:

Model of the spent fuel assembly (SFA-model) with 90 electrical heating rods



I, II, III SFA-wall temperature measuring points

⦿ 2 measuring points for central coolant temperature

Selected heating rods with temperature measuring points

Diameter of heating rod	8 mm
Length of heating rod	500 mm
Spacing ratio	s = a/R = 2,75
Number of heating rods in the SFA-model	90

Fig 2 Schematic of the SFA model including temperature measuring points

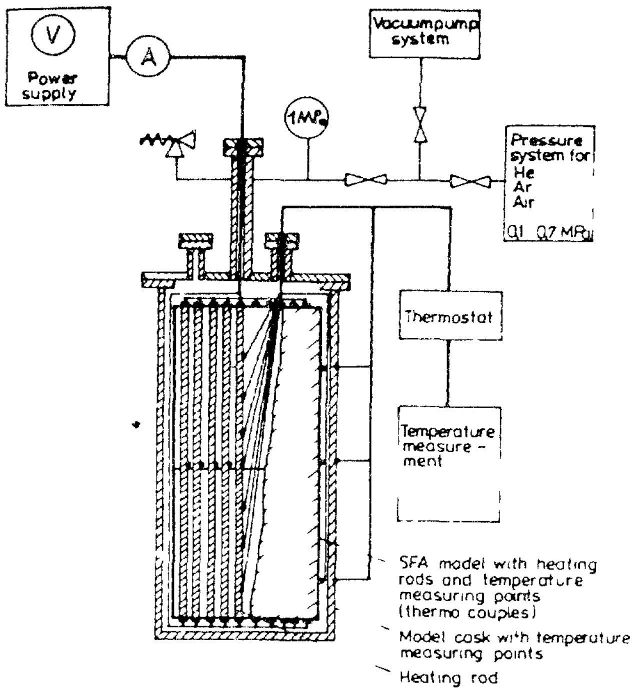


Fig 3 Schematic of the experimen al setup

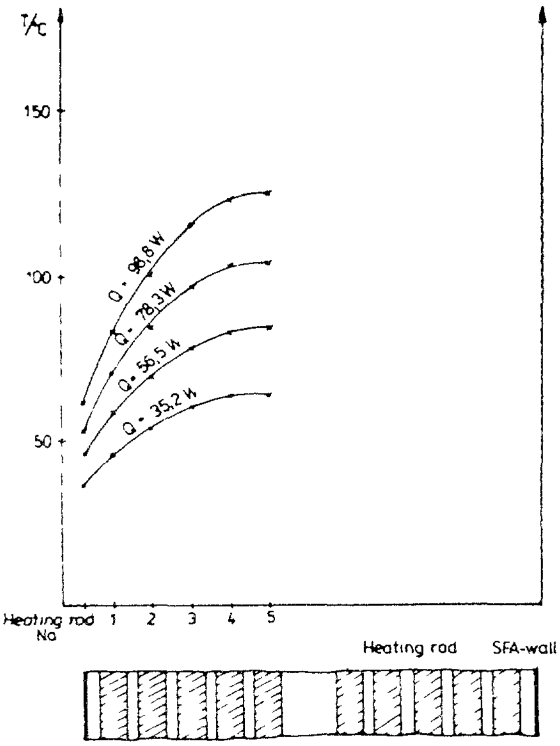


Fig 4

Radial temperature distribution of the SFA-model for different heat rates in vertical SFA-position in free air environment (ambient temperature 20°C)

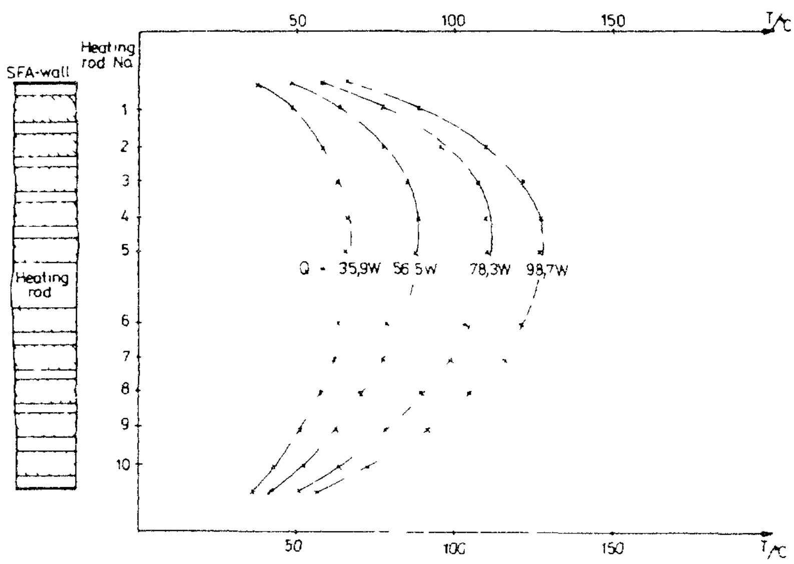


Fig 5

Radial temperature distribution of the SFA model for different heat rates in horizontal SFA position in free air environment (ambient temperature 20°C)

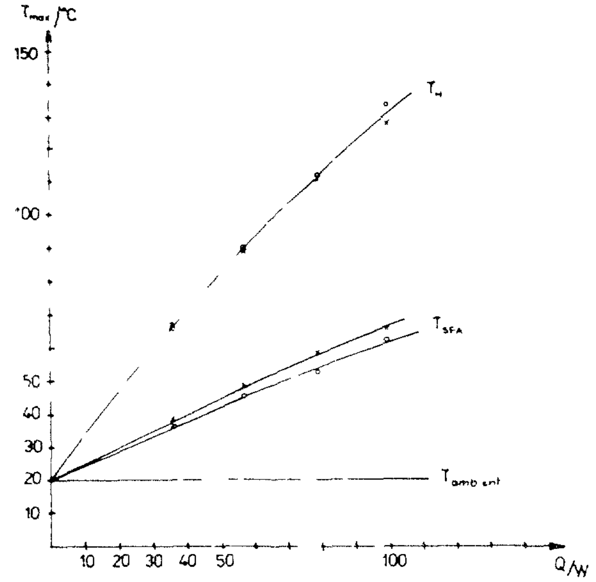


Fig 6 Maximum temperature of heating rod surface T_H and SFA wall T_{SFA} in dependence on heat rate $\dot{Q}$ for the vertical (—•—) and horizontal (—o—) position of the SFA-model in free air environment

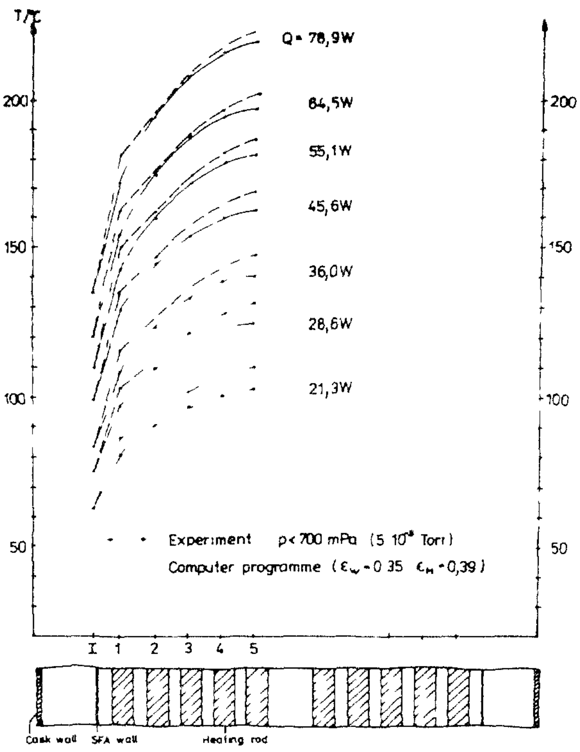


Fig 7 Measured and calculated temperature distribution under vacuum in the model cask for different heat rates $\dot{Q}$

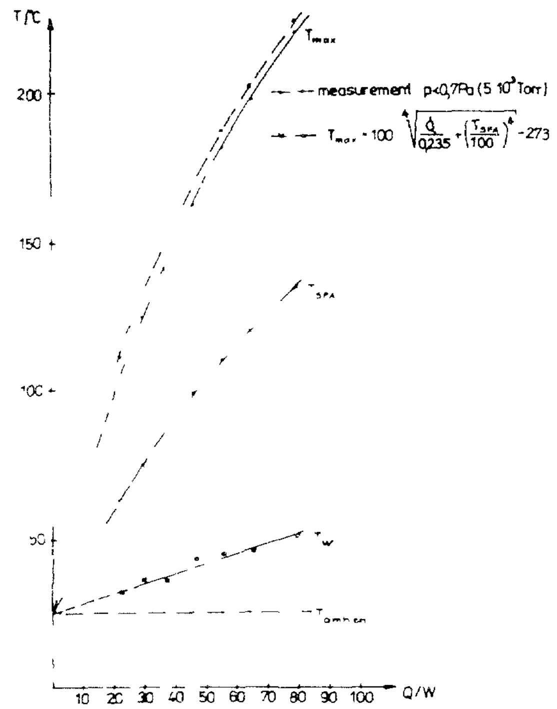


Fig 8 Maximum heating rod surface temperature in the SFA T_{max}
—•— measured under vacuum
—·— calculated for radiant heat transfer
and measured temperatures of SFA-wall T_{SFA} and cask wall T_W in dependence on heat rate $\dot{Q}$

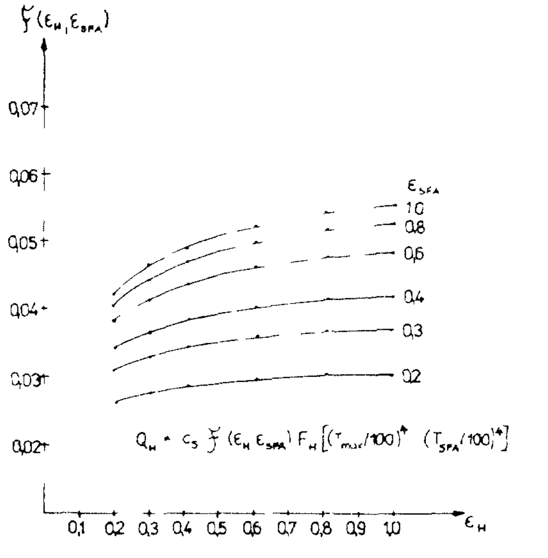


Fig 9 Dependence of radiant configuration factor $\mathfrak{F}(\varepsilon_H, \varepsilon_{SFA})$ on the heating rod and SFA wall emissivities ε_H and ε_{SFA} for the hexagonal arrangement of 90 heating rods in the SFA model with a spacing ratio s = 2,75

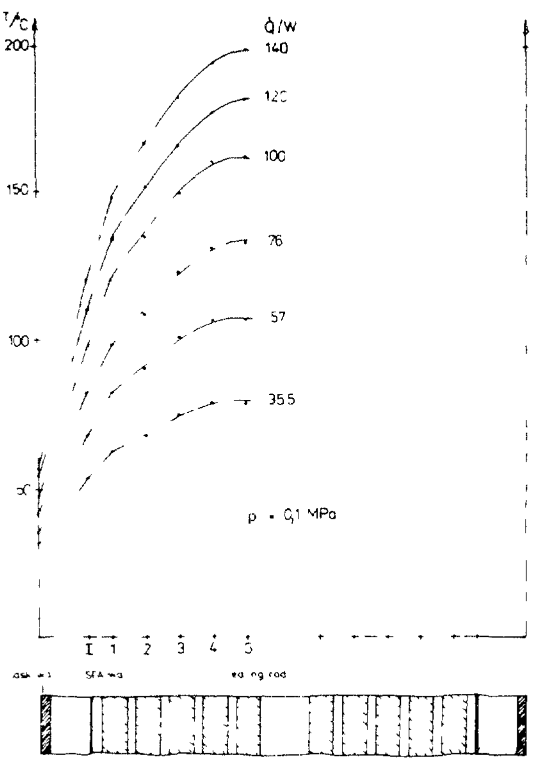


Fig 10 Radial temperature distribution of the SFA model for different heat rates under normal pressure (p=0,1MPa) in the model cask

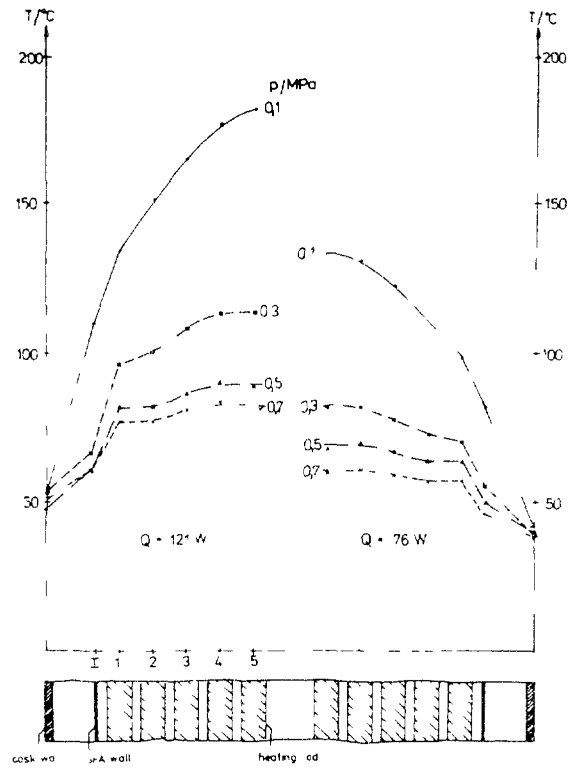


Fig 11 Radial temperature distribution of the SFA-model for heat rates of 121W and 76W and different pressure values p in the model cask

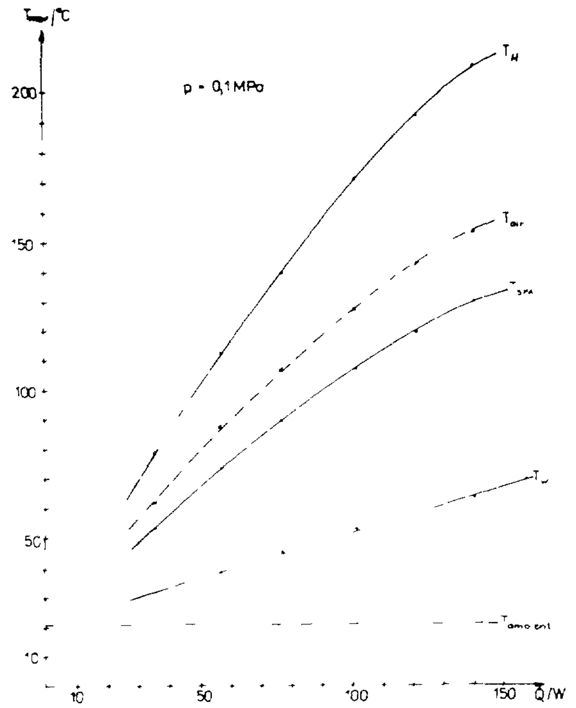


Fig 12 Maximum surface temperature of
- heating rods in the SFA T_H
- SFA-wall T_{SFA}
- cask wall T_W

and maximum air coolant temperature in dependence on heat rate $\dot{Q}$ under normal pressure (p=0,1 MPa) in the model cask

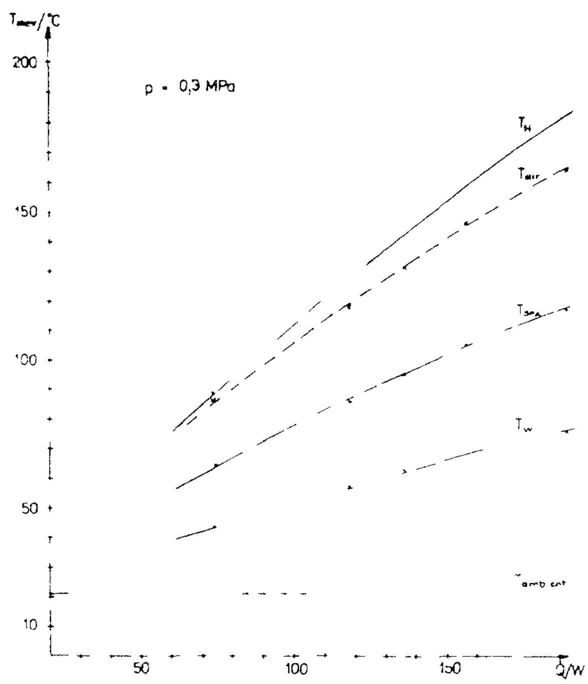


Fig 13 Maximum surface temperature of
- heating rods in the SFA, T_H
- SFA-wall, T_{SFA}
- cask wall, T_W

and maximum air coolant temperature in dependence on heat rate $\dot{Q}$ under a pressure of p=0,3 MPa in the model cask

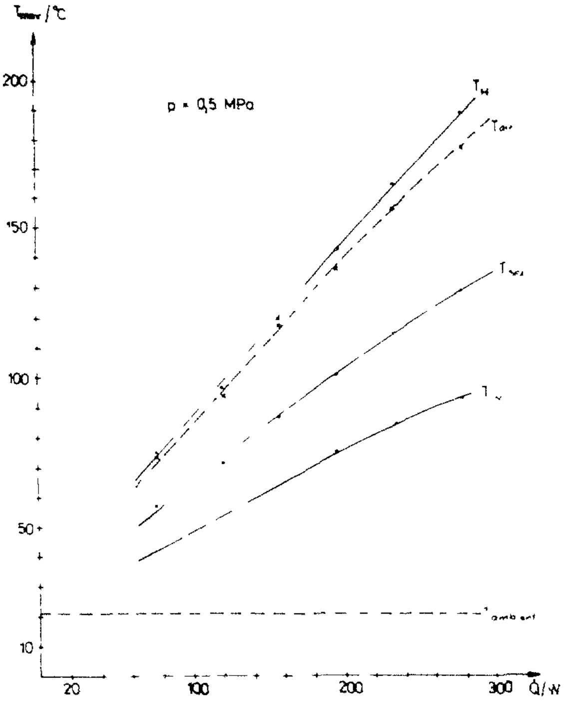


Fig. 14 Maximum surface temperature of
- heating rods in the SFA, T_H
- SFA-wall, T_{SFA}
- cask wall, T_w

and maximum air coolant temperature in dependence on heat rate $\dot{Q}$ under a pressure of p = 0,5 MPa in the model cask

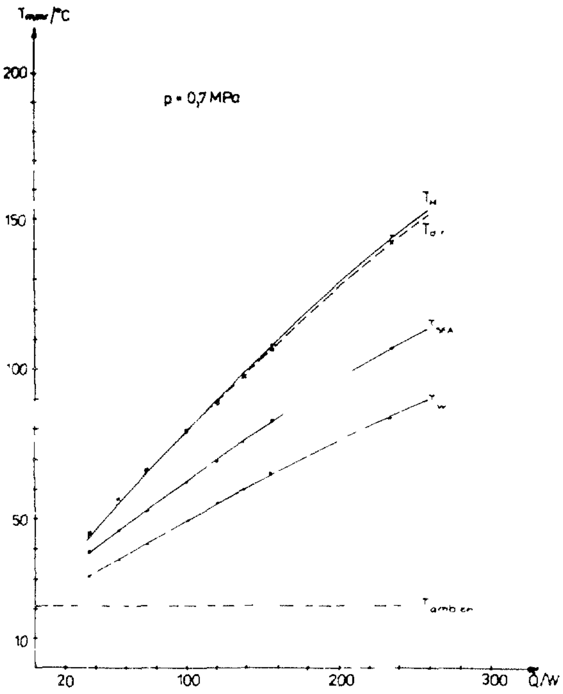


Fig. 15 Maximum surface temperature of
- heating rods in the SFA, T_H
- SFA wall, T_{SFA}
- cask wall, T_w

and maximum air coolant temperature in dependence on heat rate $\dot{Q}$ under a pressure of p = 0,7 MPa in the model cask

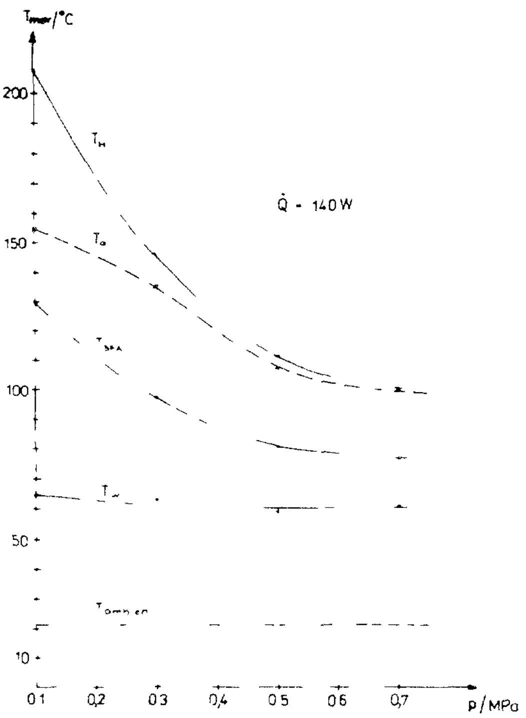


Fig. 16 Pressure dependence of the maximum heating rod surface temperature T_H in the SFA, SFA-wall temperature T_{SFA}, cask wall temperature T_w and air coolant temperature T_{air} for a heat rate of $\dot{Q}$ = 140 W

ASSESSING THE RADIOLOGICAL IMPACTS OF TRANSPORTING RADIOACTIVE MATERIALS

WORLD-WIDE RISK ASSESSMENT OF THE TRANSPORTATION OF RADIOACTIVE MATERIALS*

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1.0 INTRODUCTION

Transportation of radioactive materials is an integral part of every activity involving the use of nuclear energy, including education, medicine, industry, research and power generation. Increasing numbers and quantities of radioactive material in many different forms are being transported in the world.

The record of safety in the transportation of radioactive materials so far has been outstanding. It is believed that this is due to the fact that all shipments of radioactive materials must comply with safety standards based on or consistent with the standards recommended by the International Atomic Energy Agency in its Regulations for the Safe Transport of Radioactive Materials (Safety Series No. 6).

Although the record of safety in transport of radioactive materials is very good, public concern about safety in transport is being expressed in many countries.

2.0 PURPOSE AND BACKGROUND

The aim of the project reported in this paper is to develop the means and methods for a risk analysis of the transportation of radioactive materials throughout the world. The project was initiated by the Standing Advisory Group on the Safe Transport of Radioactive Materials (SAGSTRAM) of the IAEA. In 1979 the Swedish Nuclear Power Inspectorate and the IAEA signed an agreement on the development of a model for calculation of the transport risk. Member States of the IAEA are invited to use the model for a risk assessment of the transportation of radioactive materials in their own country. These assessments will be collected and analyzed and a world-wide risk assessment performed.

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* Research Agreement No. 2620/R1/CF.

This summary report is based upon a paper of the same title presented by Ms. Ericsson and Mr. Elert at the 7th International Symposium on Packaging and Transportation of Radioactive Material (PATRAM '83), 15-20 May 1983, New Orleans, Louisiana, USA.

The IAEA has the overall responsibility for the project and administers it. Sweden manages the project and has performed the applied research with the assistance of research support groups which have supplied data and analyses and performed some other parts of the project. An Oversight Committee with participants from eight Member States has reviewed the progress and has given valuable recommendations.

It was important that the model had the sophistication and flexibility required for its use by all Member States but still was easy to handle. The risk calculations are performed by the computer code INTERTRAN [1,2] which is based on the American computer code RADTRAN II developed by Sandia National Laboratories, Albuquerque, NM [3]. The methodology of the RADTRAN II as well as data and format of the input and output was changed to make the code more internationally oriented.

3.0 THE INTERTRAN MODEL

The computer code INTERTRAN calculates the radiological impact from incident-free transportation and from vehicular and handling accidents involving radioactive materials. The model can be divided into submodels as shown in Figure 1. These submodels are either a part of the computer code or consist of separate parts preparing input data for the INTERTRAN code.

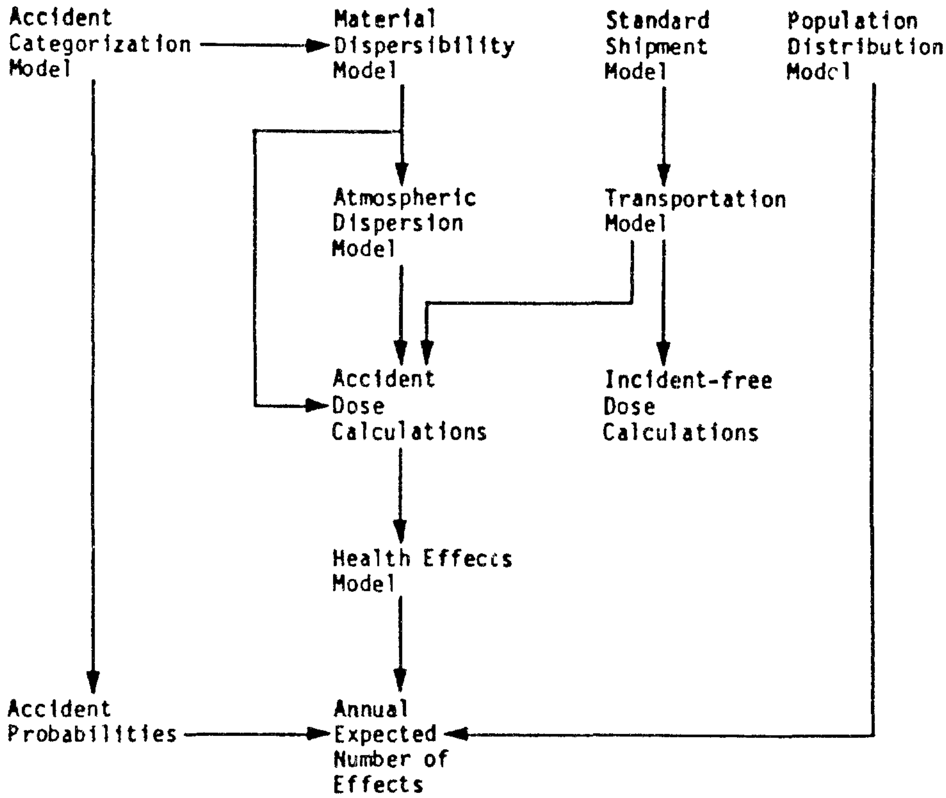

Figure 1. The INTERTRAN computer code.

A vast number of radioactive materials with varying properties are transported. Since the code cannot address all of these transports separately the Standard Shipment Model is introduced. The shipments to be analyzed are divided into standard shipments according to material, transport model, package type, etc. For each standard shipment a primary and secondary transport vehicle can be used. Up to 200 standard shipments can be calculated in one run of the code.

The Transportation model can handle ten different transport situations divided into the four major transport modes - road, rail, air and water. It consists of a traffic pattern section, a shipment section and an accident rate section.

Three population density zones with an evenly distributed population can be specified in the population distribution model. There is also a possibility to include pedestrians along the shipment path.

The incident-free dose part calculates the external radiation dose to a number of population subgroups such as:

Crew,
Passengers,
Flight attendants,
Handlers,
Population travelling on and surrounding the transport link,
Population near the transport vehicle while stopped, and
Warehouse personnel.

The impact due to incident-free transports is given as the annual expected population dose in person-rem per year.

The purpose of the accident severity categorization model is to define an accident environment or a combination of environments which give a specified impact on a package. An accident severity can be defined differently for the various transport situations but will always give the same impact on a package. For each transport situation a probability is specified for each of the eleven possible severity categories.

To describe the relative degree of damage caused by an accident to the packages of a shipment, a package failure is assigned for the accident severity categories depending on the degree of failure and the probability of failure. A different set of package failure fractions can be given for each of the ten package types.

3.1 The accident probability model

In the accident probability model the probability of a certain accident is given by:

1. The overall accident rate for the transport situation.

2. The fractional occurrence of the accident severity category for that situation.

3. The accident rate factor for the population density zone, the accident severity category and the transport situation.

3.2 The material dispersibility model

The material dispersibility model takes into consideration the dispersibility difference due to the chemical and physical properties of the materials shipped. Eleven dispersibility categories are available for classifying the materials. The dispersibility categories are assigned an aerosolization factor for each accident severity category. The aerosolization factor describes the fraction of the available material which is aerosolized and readily dispersed in an accident. Combined with the package failure fraction, the aerosolization factor gives the amount of material dispersed in an accident.

3.3 The atmospheric dispersion model

The dispersed material will be spread over a wide area and thereby be diluted. The time-integrated concentration at a specific distance from release and the size of the area with that concentration is given in the atmospheric dispersion model. The amount of material deposited in that area is also calculated in this section of the code. Dispersion data are included in the code, making it necessary only to specify the fractional occurrence of different atmospheric stabilities.

3.4 The dosimetric model

Several pathways are possible for the radiological impact on man from a transportation accident. The pathways that are included in the code are:

Internal pathways:
- Inhalation of aerosolized materials
- Inhalation of resuspended materials

External pathways:
- Ground shine
- Direct exposure from unshielded material

The inhalation dose is calculated for lung, marrow, bone, thyroid, gonads and gastrointestinal tract. The dose from direct exposure is computed using the average photon energy per disintegration for the transported nuclide.

3.5 The accident dose calculation section

In the accident dose calculation section the individual and population dose from vehicle and handling accidents are determined. The calculations are made differently for dispersible and non-dispersible materials. For the dispersible material the doses from inhalation and ground shine are calculated and for non-dispersible materials the dose from direct exposure from a damaged package is calculated.

3.6 The health effects model

The results from the dose calculations are used in the health-effect model. This section analyzes non-stochastic effects in the form of early fatalities and early morbidities and stochastic effects in the form of latent cancer fatalities and genetic effects.

The expected number of early fatalities is calculated using a dose effect table containing the probability of an early effect for a specified lung and marrow dose. For the calculation of the expected number of early morbidities, threshold values for the individual organs are used.

The expected number of latent cancer fatalities is computed as the product of the population dose to an organ and the chronic effect risk factor for that organ. In the case of non-dispersible materials, the whole-body risk factor is used.

To enable the user to do more detailed analyses of the population dose, the weighted whole-body dose is calculated from the dose to individual organs using the ICRP whole body weight factors. Two weighted whole body dose levels can be specified and the number of people receiving a dose exceeding these levels is calculated.

The risk of the transportation, presented as the annual expected number of effects, is calculated as the product of the consequences and the probability of the accident. The results are summed up for all severity categories, population zones and transport modes. Results from individual accidents can also be obtained and can be written to a separate output file for later use in calculating the cumulative probability distribution of the risk.

3.7 The handling accidents

The handling accidents are treated in a similar way as the vehicular accidents using the same accident categorization schemes. The individual dose received from a handling accident of a specific severity category is assumed to be the same as the dose received from a vehicular accident of the same suburban zone. The probability of the accident will be calculated from the number of handlings per shipment.

3.8 The sensitivity analysis

A sensitivity analysis has also been added to the code. In the incident-free case this analysis is based on the error propagation formula and calculates the relative importance of the input parameters. In the accident case the annual expected effects are divided according to population zone and accident severity category. These divided values will give the user a possibility to determine which of the parameters will cause the largest impact on the final result.

3.9 The default data sets

The code contains default data for many parameters which makes it useful for countries with different levels of data availability. The default input data concerns the transport situation, population density, accident statistics, package response to accidents, material release and dispersion, and data concerning the actual shipments of radioactive materials. At the lowest level of sophistication only shipment data are needed, although other data available will provide a more accurate analysis.

4.0 THE OUTPUT FORMAT

For each shipment the user has the option to choose between a complete output or a summary output. The complete output consists of:

For the incident-free case:

Dose to the population subgroups from the primary and the secondary transport modes.

For the accident case:

Dispersion and ground contamination values for the involved material.

Individual organ doses at different distances from an accident of each different category in each population zone.

The expected number of people in the isodose areas.

The annual expected number of early and latent effects and the expected number of accidents for each severity category.

The annual expected number of people receiving a dose higher than the specified weighted whole body dose limits.

The summary output consists of:

For the incident-free case:

The number of transported TI per year and the resulting dose for each material.

The dose to different population subgroups distributed over the materials shipped.

The dose to different population subgroups distributed over the transport modes used.

The relative importance of the input parameters.

For the accident case:

The annual expected radiological risk for each material.

The annual expected number of persons receiving more than a specified whole body dose.

The expected latent cancer fatalities for each material from ground shine, inhalation and total (sorted in decreasing order).

The annual expected radiological risk divided according to population zone and accident severity category.

A summary of the amount of material transported divided into material categories.

5.0 THE INTERACTIVE INPUT PROGRAM INREAD

To assist the user in writing the input data file, an interactive input program is provided. The program is constructed in such a way that input data can be inserted and easily combined with default data.

6.0 STATUS AS OF 1985

Following the development of the code, a detailed report and users manual was prepared and subsequently published by the International Atomic Energy Agency (IAEA) in 1983 [2]. During late 1983, the users manual was made available to all Member States of the IAEA, and it was suggested that these governments consider its usefulness for carrying out assessments of the radiological impact of transporting radioactive material. Based upon the concerns of experts however, it was not suggested that it be used for a global assessment as indicated by the title of the Research Agreement, since it was indicated that results therefrom should be used with caution "because of possible inaccuracies in the input data and intrinsic uncertainties in the methodology". During the time the code has been under practical scrutiny and in practical use the need for some corrections to the code has been identified. A meeting of interested parties was held in Stockholm, Sweden in June 1984 aiming at specifying the areas where corrections, rather than refinements, of the Code were necessary and appropriate. These corrections were introduced into the Code in late 1984.

The code has been placed in the OECD Nuclear Energy Agency Data Bank in France and it is also available from the computer code libraries located at the Argonne National Laboratory and the Oak Ridge National Laboratory in the USA. It is available in two versions -- IBM and CDC.

The code is known to be available at institutions in the following countries:

- Australia
- Austria
- Brazil
- Canada
- Egypt
- Finland
- France
- Federal Republic of Germany
- Greece
- Italy
- Japan
- Libyan Arab Jamahiriya
- Netherlands
- Philippines
- Spain
- Sweden
- Turkey
- United Kingdom of Great Britain and Northern Ireland
- United States of America

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METHODS OF ASSESSMENT OF INDIVIDUAL AND COLLECTIVE DOSES TO TRANSPORT WORKERS AND MEMBERS OF THE PUBLIC DURING THE TRANSPORT OF RADIOACTIVE MATERIAL*

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1.0 INTRODUCTION

The object of the study is to evaluate the actual radiation dose received by transport workers and collective dose to the urban public incidental to the carriage of radioactive materials. Studies elsewhere [1] suggest an empirical relation between the radiation doses received by transport workers and the transport indices of packages handled.

The investigation reported here relates to the transportation of radioactive materials in Type A packages for use in medicine, industry and research. Such radioactive materials are supplied by the Bhabha Atomic Research Centre, Bombay to a large number of users all over India.

2.0 INDIVIDUAL DOSE RECEIVED BY TRANSPORT WORKERS

Handlers of radioactive material packages are generally exposed to the highest dose rates of any population group. However, the actual doses received tend to be relatively small because of the short period of exposure [2]. For determination of the dose received by transport workers, individuals who were engaged in the handling of radioactive cargoes at the Bombay Airport were provided with personnel monitoring badges. However, with the low dose recorded by the badges, it was not easy to develop an empirical model for estimating the individual doses. For our purpose, it was necessary to identify individuals who were (a) engaged in handling radioactive cargoes and (b) likely to receive measurable amounts of radiation dose in the course of handling packages. Such persons were identified at the Bhabha Atomic Research Centre itself from the group of individuals who were engaged in packing the consignments and forwarding them for transport. These individuals were designated as transport workers specifically for the purpose of the investigations.

These "transport workers" are routinely engaged in work with radiation and are regularly monitored. Each of these persons was provided with an additional personnel monitoring badge and was instructed to wear it only when he was carrying out any work connected with transport of radioactive materials. These additional badges were replaced generally fortnightly and the dose received by each of them was separately recorded.

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2.1 Data regarding packages

In order to be able to correlate empirically the radiation dose received by individuals to the volume of consignments handled, the following information was required:

- The number of packages handled
- The nature of the radioactive contents
- The design description of the package
- The radiation levels outside the package.

The packing note which is prepared for every package despatched is a source of valuable information regarding the consignment.

Upon examining individually well over 11,000 packing notes, it was clear that only packages containing iodine-131, molybdenum-99 and iridium-192 industrial radiography sources would contribute significantly to personnel dose. The radionuclides are transported in packagings of standardized design. The choice of the packaging depends on the source strength and the shielding required.

Typically the packages of interest for dose assessment viz. those containing iodine-131, molybdenum-99 and iridium-192 constitute 60% of all the packages despatched. About 22% of the total number of consignments are exempt packages.

Of the remaining packages about 55% consist of pure beta emitters, such as phosphorous-32 and sulphur-35, which are despatched in small quantities in category I-WHITE packages. What remains would effectively account for 30 GBq (800 mCi) per month made up of 25 radionuclides despatched in 50 packages.

The most frequently transported consignments of interest totalling 12 in number were identified and these were supplied in specific denominations of activity including 4.6×10^{-3} GBq (125 μCi) of molybdenum-99 and 30 GBq (8 Ci) of iridium-192. The radiation levels at specific distances from the different consignments were measured. The values corresponding to the twelve consignments are shown in Table 1.

2.2 Work study

A detailed work study was carried out to determine the distance between the persons and the packages in every stage of handling and the corresponding period of handling. There were several stages of handling the packages such as collecting individual lead pots containing the source, packing the lead pot in an outer container, labelling the package and forwarding it for transport. The observed times and distances fell within a narrow range of values, owing probably to the experience gained by the workers in handling thousands of packages, and are given in Table 2.

2.3 Calculation of individual dose

Ten stages of package handling are listed in Table 2. Let "k" denote the stage number. Individual packages are handled in stages k = 1 to 4, 6 and 8. Operations 5, 7, 9 and 10 involve handling of all packages together.

The total dose per despatch in handling all the consignments over the various stages is:

$$XH = \sum_{j=1,12} N_j * \Big(\sum_{\substack{k=1,8 \\ k \neq 5,7}} X_{j,k} * t_k\Big)$$

where N_j is the number of consignments of index number j, X_{jk} is the dose-rate at distance r_k and t_k is the period of handling in stage k. The values of r_k and t_k are given in Table 2.

The above relation for XH takes into account only the operations involving handling of individual packages. In the case of stages 5, 7, 9 and 10 the persons who were engaged in the job were provided with personal dosimeters and the dose per person per despatch was obtained. Adding this to XH, the assessed dose received by individuals per despatch was computed and designated as XA.

2.4 Results and discussion

The assessed dose, XA, for a given period computed on the basis of the above empirical model was compared with the recorded doses, XR obtained from the personal dosimeters worn by the personnel during the same period in Table 3. It was noted the XA $\approx$ XR within $\pm$ 25% in most of the cases [3].

The above model was then used to assess the doses received by the package handlers at the Bombay airport. This work practice was characterized by d = 20 cm and t = 100 sec per package. The value of "t" was large because the transport workers remained at the relevant distance handling many non-radioactive consignments. The values of the assessed dose equivalent received by the cargo handlers, XB, estimated on this basis are listed in Table 4 for five typical assessment periods.

The third column of Table 4 suggests that the dose equivalent received by the transport workers at the airport could also be calculated, independent of the above model, based on the total transport indices handled. Thus:

$$XB = 2.7 \times 10^{-6} \times (\textstyle\sum_j (TI)_j * N_j) \text{ person-Sv}$$

where $(TI)_j$ is the transport index of the consignment type j, N_j is the number of consignments of type j, and 2.7×10^{-6}, is an empirical quantity which is the average of the ratio values, shown in the third column of Table 4.

The assessed doses between a method based on the above model and an empirical relation from the results of Shapiro's study [1] are in agreement. This gives confidence in using the above model for the estimation of doses to airport transport workers.

The above estimate, XB, represents the collective dose received by the transport workers at the Bombay airport. The consignments fan out to different destination airports all over India. At these airports, the packages are collected by radiation workers from the consignee institutions. The collective dose (i.e. person-Sv) received by cargo handlers in these airports together should be nearly equal to the corresponding XB because (a) it was noted that the time and distance factors applicable to the unloading operation were not significantly different from those for the loading operation, and (b) at the Bombay airport fewer persons received more millisievert while at the other airports in India more persons receive fewer millisieverts each. Yet a correction is to be made in respect of the molybdenum-99 consignments.

On account of the radiological decay of molybdenum-99, the change in the energy profile, there will be a fall in the exposure rate outside the packages at the destination. Our computations show that in a given period, the collective dose to the cargo handlers at the destination airports in India would be 8% less than XB.

The results obtained on the basis of the above model are, naturally, applicable to the local conditions specified in the study. However, the model is generally applicable to any situation. At any airport the dose to the cargo handlers can be roughly estimated, given the transport indices of the packages handled. A simple empirical method of dose assessment is offered by this model, based purely on the exposure rates at the relevant distances from light packages containing the commonly used gamma emitters in specific denominations of activities, the number of packages handled and the time and distance factors involved in package handling.

For the movement of more than 11,000 packages of radioactive materials during the period 1982, from Bhabha Atomic Research Centre, Bombay, India, through the Bombay airport and through distribution airports throughout India to their final destinations, the estimated annual collective dose to transport workers is 0.0142 person Sv (1.42 person rem). Most of this exposure results from packages containing iodine-131, molybdenum-99 and iridium-192 industrial radiography sources and is incurred by baggage handlers at airports.

3.0 COLLECTIVE DOSE TO URBAN PUBLIC

For estimating the collective dose to members of the public we identified the persons who were likely to be exposed to radiation and measured for radiation levels at the locations occupied by them. For the type of consignments, it was possible to develop a model for collective dose estimate laying emphasis on pedestrian density along the route. The pedestrian density could be assigned a constant value along the route as is the common practice in making such estimates [4]. However, in view of the sharp variations in the pedestrian density along a route within the city, the routes were divided into a number of segments such that the density of exposed persons within a segment was approximately uniform.

The vehicle takes radioactive cargo to the airport and to a local source collection centre through residential areas, busy traffic-ridden roads, crowded bazaars and highways. In certain places the pedestrian density is usually high, while in others those waiting at bus stops provide the population density. Also to be considered in estimating dose to public is the passenger density in vehicles around the delivery van.

On the relevant routes, there are essentially three lanes of traffic in each direction. The vehicle carrying the radioactive consignments drives either along the extreme left or the middle lane but rarely in the right lane.

For assessing the radiation levels to which the pedestrians may be exposed, the area around the vehicle was divided into a number of cells, each cell being a 1 m x 1 m square which could be occupied by an individual. The radiation level in each cell was measured and the average of a number of observations obtained. In order to take into account the partial shielding provided by those closer to the vehicle, the radiation levels in various cells were measured when persons were present near the vehicle for brief durations, engaged in such work as supervising the loading operations and checking the vehicle. Thus the radiation levels around the vehicle were mapped for two configurations [5] viz., (i) the van occupying the left lane and (ii) the van occupying the middle lane. The distance over which the pedestrian density was nearly uniform was labelled as a segment with the population density being different in the different segments.

Each segment can be divided into identical sub-segments. A sub-segment is that length which is covered by a configuration. The pedestrian density in a sub-segment is independent of the vehicle configuration. However, the passenger occupancy distribution depends on the configuration. Thus for each sub-segment we have a 16 x 3 pedestrian occupancy matrix. The side walk can be divided into three columns running parallel to the road.

3.1 Details of calculation

As the vehicle moves along the road, the radiation level at each cell increases and then decreases in a manner characteristic of the configuration. Each cell is irradiated for the same duration for every position of the advancing vehicle. The length of a cell along the direction of motion is 1 m. Given the speed of the vehicle, the period of exposure, Δt could be determined. As the vehicle approaches and moves past, every person in column j would receive a total dose given by

$$D_{j,l} = (\Sigma_i \; X_{i,j})_l \Delta t$$

where $X_{i,j}$ is the dose rate at the cell represented by row i and column j of the configuration, l.

If there are a total of n_j persons in the column j over the entire length of a segment k, the collective dose received by all these persons is given by

$$D_{ped}(k) = \Sigma_l \; (\Sigma_j \; n_j \; D_{j,l})$$

Thus the total dose received by pedestrians on all the segments in a route is given by

$$\Sigma_k \; D_{ped}(k)$$

Dose to passengers in the nearby vehicles may be calculated easily as the passenger distribution is independent of the segment. Thus, the dose to passengers in route 'r' is given by

$$D_{pass} \; (r) \;\; = \;\; (\sum_{j > 3} \; n_j \; (\sum_i X_{i,j})_l \; \Delta t_r)$$

Since the lengths of the routes are known and the speeds of vehicles are equal, the period of exposure Δt_r is easily computed.

The estimated annual collective dose received by public can be obtained as

$$CD_{Ann} \;\; = \;\; \Sigma \;\; CD_r \; x \; N_r$$

where N_r is number of trips along the route 'r' per year and CD_r is the collective dose per trip in route r being equal to

$$\sum_k D_{ped} \; (k) \; + \; D_{pass} \; (r)$$

the summation in the first term, here being over the number of segments in route 'r'.

3.2 Results and discussion

On the basis of observed pedestrian density in different segments along the routes, and making use of the values of the radiation levels, $X_{i,j}$ for the two configurations, the estimated annual collective dose resulting from transportation of radioactive consignments is about 0.1 person-Sievert (10 person-rems). In India, the dose to public due to urban transport of radioactive materials is likely to be significant only in Bombay where all these consignments originate. After despatch, these packages get distributed over the different towns in India so that in a given town fewer packages would be transported in several vehicles belonging to individual users.

In the above collective dose estimate for Bombay, while the radiation levels at the various cells are low, the above collective dose results from the relatively high pedestrian/passenger density. Actually vehicular configurations other than those assumed here are possible. The passenger density in the neighbouring vehicles may vary and it is the simplifying assumptions which make this collective dose estimate model manageable, although observed variations in the pedestrian density are not so large as to significantly affect the results.

The advantage offered by this model is that the dose to public could be optimized by judicious choice of the route, time of transport and configuration. In our computations it was noted that the dose to public would be minimal if the delivery van took on the configuration 2 in high pedestrian density areas and the configuration 1 in other areas. The above model is generally applicable to any urban environment given the pedestrian densities along the route.

In a given segment of the route the pedestrian density may display fluctuations but if a segment is divided into a number of bits, these fluctuations would cease to be important, since the pedestrian density is characteristic of the place for the given time of the day. However, this division and sub-divisiion of segments may render the computation more tedious than could be justified by the improvement in the reliability of the calculated result.

On the other hand, the pedestrian density could be averaged over the entire route and with a simple interaction of a single average pedestrian density matrix and a radiation level matrix, the collective dose could be obtained. Such simplification may be justifiable in computations involving highways running through small towns and rural areas where the population density per unit distance (say, 50 or 100 km) may be nearly constant. In urban environments, particularly where the relevant route passes through areas of sharply varying pedestrian densities, it may be worth adopting segmental dosimetry in order to decide upon the route, time of transport and the configuration for optimizing the dose to public. However, if the configuration is forced by the traffic pattern, then a computation involving an interaction between the radiation level matrix and a single average pedestrian density matrix obtained by giving due weighting to the pedestrian densities in the different segments, may be simple, acceptable and general in nature.

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TABLE 1

Radiation Levels at Different Distances from the Consignments

Consignment Index	Dose rates at specified distance (cm) from the consignment ($X\ 10^{-2}$ mGy/h or mrad/h)						
	0	10	20	30	40	50	100
1	5.2	1.9	1.1	0.7	0.4	0.3	-
2	8.6	2.3	0.9	0.7	0.4	0.3	-
3	45.6	12.3	6.3	4.0	2.5	1.8	0.4
4	5.1	3.0	2.3	1.6	1.1	0.8	0.6
5	38.0	11.0	5.9	3.5	2.5	1.5	0.5
6	9.2	3.8	2.7	1.8	1.3	0.7	0.2
7	18.0	8.0	5.0	2.5	2.0	1.5	0.8
8	50.0	15.0	9.0	7.5	6.5	5.5	1.6
9	60.0	30.0	15.0	9.0	5.3	2.6	0.7
10	120.0	42.0	22.0	11.0	7.0	5.0	1.5
11	12.0	4.5	3.6	2.7	2.0	1.5	0.7
12	9.0	4.0	3.5	2.8	2.3	1.5	0.2

TABLE 2

Distances at which packages are handled and the corresponding duration in the handling of packages

Operation sequence number (k)	Details of operation	Distance between person and package r_k (cm)	Duration of operation t_k(sec)
1	Taking single lead pot for packing	10	5
2	Sealing the lid of the lead pot	20	30
3	Packing the lead pot in an outer container	30	100
4	Affixing labels	10	40
5	Storage of packages behind a lead castle	50	3600
6	Transfer of package to a push cart	20	5
7	Shifting of loaded push cart to despatch unit	30	100
8	Loading of individual package in the delivery van	20	5
9	Journey to the delivery point	300	2700
10	Handing over packages at the delivery point	300	3600

TABLE 3

A comparison of the assessed dose and the recorded dose

Monitoring Period No.	Duration of Monitoring (days)	Assessed Dose (XA) ($X10^{-5}$) person Sv	Recorded dose (XR) ($X10^{-5}$) person Sv	Percent Difference (XA XR)/XR x 100
1	14	351.6	485	-27.5
2	14	349.2	284	+23.1
3	14	363.6	254	+43.1
4	14	365.8	470	-22.1
5	30	732.2	510	+43.0
6	18	423.6	552	-23.3
7	14	350.9	315	+11.4
8	10	269.3	409	-29.3
9	23	375.6	304	+23.6

TABLE 4

The Assessed Dose Equivalent (person-Sv) Received by Transport Workers at the Bombay Airport

Assessment Period No.	Assessed Dose XB ($X 10^{-5}$ person Sv)	Ratio = $\frac{XB}{\sum_j (TI)_j N_j}$ ($x 10^{-5}$)
1	30.0	0.27
2	27.6	0.31
3	34.3	0.33
4	35.2	0.34
5	28.5	0.34
6	32.4	0.29
7	34.2	0.32

STUDY ON THE RADIATION EXPOSURE OF TRANSPORT PERSONNEL*

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1.0 INTRODUCTION

The study had the following objectives:

- Exploitation of the dosimetry results for employees responsible for handling and transporting radioactive materials, radiopharmaceuticals, spent fuel and other radioactive material;

- Evaluation of the corresponding collective detriment;

- Formulation of recommendations for optimization studies aimed at reducing these radiation exposures.

In view of the confidential nature of the data and the fact that the transporting companies are scattered, this study could only cover a small number of facilities and carriers. Owing to the various types of transport carried out by the transporting companies, it was often not possible to differentiate between types of transport and between the doses resulting from handling and those resulting from the actual transport.

The results supplied below cover three firms which, for the purposes of the study, we shall call A, B and C.

Firm A dealt almost exclusively with the conditioning, transport and delivery of radiopharmaceuticals or small sources used in nuclear medicine.

Firm B dealt with various types of transport: uranium, plutonium, fresh and spent fuel elements and waste. Overall results are given but a distinction is made between doses received during transport and those arising from handling.

Firm C dealt mainly with the transport of nuclear materials (plutonium and enriched uranium).

2.0 RESULTS OF FIRM A

2.1 Products

The marketed products are divided into three main categories:

- Radiopharmaceutical products used mainly for in vivo diagnosis and radiotherapy; these are all transported in Type A non-returnable packagings (from one millicurie to several hundred millicuries);

* Research Agreement No. 2792/R2/CF.

- Radioactive sources to be used either in nuclear medicine (radiotherapy or curietherapy) or in industry; these are transported in Type A and Type B returnable containers.

  Type A containers are used for transporting medium-level sources.

  Type B containers are used for transporting high-level and very high-level sources.

- Products for medical analysis to be used for in vitro diagnosis; these are transported in the form of low-level reagent kits (a few millicuries) packed in cardboard and in non-returnable Type A packages.

The number of packages delivered, using 1980 as a reference year, was as follows:

	Number of packages		Proportion %	
	Type A	Type B	Partial	Total
- Non-returnable Type A packages (expendable packaging)	52 700		94	
- Returnable quality Type A casks	3 000		5.5	
Total Type A packages	55 700			99.5
- Returnable heavy shielded Type B casks		300		0.5
Total "labelled packages"	56 000			100%

The only packages capable of causing irradiation of personnel belong to the category of "labelled active packages" (56 000). In reality these consist of the first two types: radiopharmaceutical products (52 700) and radioactive sources (3000 + 300), in particular non-returnable Type A packages (standards and $^{99}Tc^{m}$ generators).

2.2 Radioactive materials marketed - number of packages delivered - annual round trips made by vehicles (reference year 1980)

Radioactive materials marketed in the form of radioisotopes

Standard boxes	Iodine-131	~ 760 Ci
	Other radiopharmaceuticals	~ 410 Ci
$^{99}Tc^{m}$ generators		~ 6000 Ci

Number of packages delivered

Standard boxes ~ 25 200 including ~ 12 500 for iodine-131.
$^{99}Tc^{m}$ generators ~ 27 500 including ~ 14 000 for EEC countries, which are not considered as daily transports.

Annual round trips made

Eight vehicles go twice a day to:

Paris and the Paris region)		
Airports)	Delivery	~ 38 200 packages
Railway stations)		

Thus, with an annual rate of ~ 4000 trips (250 working days x 16), each of these drivers is responsible for an average of 9-10 radioactive packages per day, divided up as follows:

6-7 standard packages, including 3 of iodine-131;
3-4 $^{99}Tc^{m}$ generators;

in other words, an average daily transported radioactivity of:

3 x 60 mCi	~ 180 mCi of iodine 131
4 x 34 mCi	~ 136 mCi of other radiopharmaceuticals
3-4 x 450 mCi	~ 1350-1800 mCi of ^{99}Mo-^{99}Tc.

All these packages are subject to rules, codes and national and international agreements. Their dose rate always meets the prevailing norms, viz.:

200 mrad/h in contact   10 mrad/h at 1 metre.

2.3 Transport of packages

This has a number of characteristics which are related both to technical constraints imposed by the facilities and to constraints associated with delivery. Transport obviously involves the following actions: preparing the package, collecting it, transporting it and delivering it. Each of these actions, in the case of "radioactive packages", is a potential source of radiation exposure to the employees performing them.

The packages are "prepared" by a team of nine (eight until 1980) in the dispatch hall.

The packages are loaded, transported and delivered by a team of ten drivers, eight of whom are on duty. Thus, the damage resulting from "transport" is the capitalization in man-rem of the radiation exposure incurred in the performance of these various tasks. Each employee carries a film dosimeter at all times during working hours.

Hence it is worth examining the radiation to which the above employees are exposed and identifying the damage arising from each of the four stages mentioned above.

2.4 Radiation exposure during preparation of packages for dispatch

From the moment it arrives from the production laboratories until the final packing stage, a standard box passes through a number of work posts. Until the end of 1982 the arrangements were such that manual intervention was necessary at most of these work posts, which thus gave rise to radiation exposure of the employees working there.

For the years 1978-1981, the record is as follows:

Dispatch hall	1978	1979	1980	1981
Radiation exposure in man-rem	~ 13 (8 employees)	~ 11 (8 employees)	~ 12.5 (8 employees)	14 (9 employees)
Packages prepared for Paris-airports-stations	~ 30 500	~ 34 000	~ 38 200	~ 57 000

which corresponds to an average monthly exposure for each employee of ~

135 mrem in 1978
114 mrem in 1979
118 mrem in 1980
130 mrem in 1981

directly associated with handling (for 1980 as an example) of:

3 x 23 working days = 69 packages of iodine-131
4 x 23 working days = 92 packages of other radiopharmaceuticals
3-4 x 23 working days = 69-92 $^{99}Tc^{m}$ generators.

2.5 Exposure incurred during collection - transport - delivery

These operations are performed by the drivers. Irrespective of its destination (nuclear medicine service, research laboratories, railway stations, airports), each package is loaded into its vehicle, transported and delivered. All these operations are performed manually and, as in the case of preparation of the packages, lead to exposure of the drivers.

For the years 1978-1981 the record was as follows:

Drivers	1978	1979	1980	1981
Radiation exposure in man-rem	~ 13.2 (8 drivers)	~ 9 (8 drivers)	~ 10 (9 drivers)	11.5 (9 drivers)
Packages transported Paris-airports-stations	~ 30 500	~ 34 000	~ 38 200	~ 57 000

which corresponds to an average monthly exposure per employee of ~

137 mrem in 1978
94 mrem in 1979
93 mrem in 1980
106 mrem in 1981

directly associated (for 1980 as an example) with handling during the above-mentioned operations.

2.6 Comparative records

(a) It seemed useful to establish the difference between the exposure incurred during the various handling operations and that incurred during the actual transport, although it must be made quite clear that the principle should be accepted that all these operations are part of "transport", even if it is hardly possible to do anything about the problems which may arise during delivery.

For this reason, in December 1981, three drivers were provided with extra film dosimeters to be carried only while loading the packages into their vehicle and while unloading and delivering them. The results are significant and demonstrate clearly that the exposure incurred during handling is similar to that during the actual transport.

Moreover, it gives a valuable indication of the effectiveness of the protection installed in the vehicles: for a daily average of three hours spent in his vehicle, each driver is exposed to radiation of the order to 2-4 mrem, which is within the acceptable limits.

Drivers*	Personnel dosimeter	Extra dosimeter
1	350 mrem	150 mrem
2	180 mrem	75 mrem
3	230 mrem	130 mrem

* 2 journeys per day, 3 hours spent in driving.

(b) The total radiation to which the employees are exposed during the four reference years is now examined.

1978: 16 employees (8 in hall + 8 drivers) took part in the packing, transport and delivery of 30 500 "radioactive packages" and accumulated 26.2 man-rem, or an average of 1.630 rem/employee, corresponding to a daily average exposure of 6.5 mrem for the transport of 7-8 packages.

1979: Also 16 employees, but 34 000 radioactive packages transported and 20 man-rem accumulated, which is an average of 1.250 rem/employee, corresponding to an average daily exposure of 5 mrem for the transport of 8-9 packages.

1980: 17 employees (8 in hall + 9 drivers) dealt with 38 200 "radioactive packages" and accumulated 22.5 man-rem, or an average of 1.32 rem/employee, corresponding to an average daily exposure of 5.6 mrem for the transport of 9-10 packages.

1981: 17 employees dealt with about 57 000 packages transported and received an accumulated dose of 25.5 man-rem or an average of 2 rem/employee, corresponding to a daily average exposure of 7 mrem for the transport of 12 packages.

Analysis of the following table leads to two observations:

	1978	1979	1980	1981
Number of packages per day	7/8	8/9	9/10	12
Daily exposure in mrem	6.5	5	5.6	7
Exposure per package in mrem	0.86	0.58	0.58	0.58

- The average daily exposure of each of these employees is far below the legally permitted level of 20 mrem;

- The average daily exposure of the employees in 1979 decreased from 6.5 mrem to 5 mrem whereas the number of packages transported increased by 10%. This reduction is connected with the first stage of a reorganization of the dispatch hall involving an increase in the useful surface area by adding an outside canopy and a walkway to the first floor reserved for Sales Administration;

- Research is being carried out continuously to reduce exposure of personnel still further, taking into account the steady increase in the sale of radioisotopes.

  This research has led to the complete reorganization of the dispatch hall.

This extensive rearrangement has two objectives. The first is to improve the technical resources, as these are no longer adequate to meet the needs engendered by the steady increase in sales (of the order of 10-15 [%] per year in volume), which by 1983 should result in the dispatch of 100 000 packages, of which 65 000 will be radioactive.

The second is to reduce the radiation exposure of employees working in the hall by making maximum use of mechanization through a combination of techniques and methods; it is hoped that the total exposure of employees will be reduced from the present figure of approximately 12 man-rem to a maximum of 2 man-rem. The first part of the reorganization has been in operation since October 1982. It consists of the automatic conditioning line for small Type A packages.

- A similar type of line but adapted to ELUMATIC packages ($^{99}Tc^{m}$) is being installed and will be put into operation at the end of 1983.

- The new automatic conveyor for distributing all the packages to the loading bay will become operational at the end of 1984.

From this date onwards only the actual loading into the vehicles will be done manually.

During the whole of this traditional period, the doses received will be monitored continuously and this will provide accurate information on the benefits of the reorganized system.

However, it should be recalled that a cost-benefit analysis as defined in the recommendations of ICRP Publication 26 (optimization principle) gives a monetary cost α equivalent to the damage caused by irradiation of the order of 1 million French francs per man-rem.

- Production development:

This concerns the putting into service of a cyclotron for the production of short-lived, low-energy radionuclides which nuclear medicine services will be using more and more frequently.

As a result, radioactive packages to be used for the same purpose will have much lower dose rates than those marketed at present and will therefore be less harmful for the staff.

- Unfavourable factor:

The most unfavourable aspect, and one which is virtually inevitable, concerns the transfer of air transports from Orly to Roissy-Charles de Gaulle, which is already in progress. The transport time, and hence the exposure time, will be considerably increased. It should be recalled that nearly 60% of the transports are by air.

2.7 Conclusions for Firm A

It must be recognized that as a result of precise and comprehensive regulations which are strictly observed, the transport of radioisotopes does not in itself present any real hazard either to the environment and to members of the public - which is the most important thing - or to transport personnel. Nevertheless, this firm constantly endeavours to improve the working conditions and methods as far as possible in order to reduce to the lowest possible threshold any damage resulting from ionizing radiation. In doing so, they have even accepted a higher cost-effectiveness ratio than can be reasonably expected at the present time. The transport of radionuclides is, however, unquestionably justified since nulcear medicine and its therapies depend on it.

3.0 RESULTS OF FIRM B

Firm B carries out:

- The transport of PWR or BWR spent fuel packages by road in Type B packaging;

- The transport of various radioactive materials (radioisotopes, wastes, fuel assemblies for research, irradiated samples and so on) in all types of packaging (industrial, A, B);

- The handling and monitoring of these radioactive materials during transport.

For the years 1980, 1981 and 1982 the following results were obtained (although the data supplied are less precise, we have tried to list separately the doses received during handling and those received during transport):

Type of transport	Number of employees	Average individual doses (mrem)	Collective dose (man-rem)
<u>1980</u>			
Spent fuel	7	0	0
Various materials	15	23.3	0.34
Handling and monitoring	1	0	0
<u>1981</u>			
Spent fuel	7	0	0
Various materials	15	12	0.18
Handling	1	0	0
<u>1982</u>			
Spent fuel	7	0	0
Various materials	15	12	0.20
Handling	1	0	0

These results show that, as in the previous study, it is the various radioactive materials which lead to the highest exposures, although they are still very small. Exceptionally, handling and monitoring do not seem to give rise to any exposure, but the result is not significant since only one employee was involved.

It may be assumed that the drivers who took part in the transport of the "various materials", including the minor activities delivered, were exposed in conditions similar to those of the delivery group in Firm A.

4.0 RESULTS OF FIRM C

Firm C transported mainly nuclear materials: natural uranium, enriched uranium, plutonium in metal and oxide form.

Six drivers carry out these transports, which involve a total of about 250 000 km covered annually, of which 30% is accounted for by enriched uranium, 20% by plutonium and PuO_2, 4% by UO_2 and 40% by the return of empty packagings.

Of these transports, 20% are made with only one driver and 80% have a team of two drivers, which is an average of 72 000 km per driver per year.

The drivers carry a film dosimeter all the time and measurements are also made in the cabs.

Measurements taken in the cab before departure give zero radiation levels. It may be concluded that the exposures are due mainly to handling of packages during loading and unloading or even to prolonged waiting in a hot laboratory near a source before loading.

Systematic accurate neutron measurements have been made for a year and have always given zero values both in the cab and on the films.

- In summary, the annual average radiation exposure of the employees involved in transport is:

25.3 mrem for the skin with a maximum of 50 mrem for one employee
25.3 mrem for the whole body
18.3 mrem for the wrist

The average time spent by the employee in driving vehicles is about 1200 hours.

	CUMUL. 12 MONTHS									CUMUL. 36 MONTHS			ANNUAL AVERAGE			AVERAGE man/yr		
	1980			1981			1982											
EMPLOYEE	S*	WB*	W*	S*	WB*	W*	S*	WB*	W*	S*	WB*	W*	S*	WB*	W*	S*	WB*	W*
A	0	0	0	0	0	0	130	130	75	130	130	75	43.3	43.3	25			
B	0	0	0	0	0	0	35	35	75	35	35	75	11.6	11.6	25			
C	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0			
D	0	0	0	35	35	0	0	0	0	35	35	0	11.6	11.6	0	25.3	25.3	18
E	35	35	0	75	75	75	0	0	0	110	110	75	36.6	36.6	25			
F	35	35	35	115	115	35	0	0	35	150	150	105	50	50	35			

*S - skin
*WB - whole body
*W - wrist

As for the transport of PuO_2 from LWRs, specific measurements were made to ascertain the doses:

- Calculations and experiments:

Experiments with FS 51 cages (8 kg of PuO_2) on given geometries representative of a vehicle load make it possible to determine the type and thickness of shielding which must be installed in the vehicle in order to protect the cab.

For one year, drivers were supplied with thermoluminescent neutron dosimeters, and dosimeters were also installed in the cab. Zero doses were recorded. Following these results, it was decided to stop using the thermoluminescent film dosimeters.

DEVELOPMENT OF GENERIC RISK ASSESSMENT CODE FOR RADIOACTIVE MATERIAL TRANSPORTATION*

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1.0 INTRODUCTION

The research agreement related to three basic tasks in the area of risk assessment; these included support in development of the INTERTRAN code, development of risk assessment methods and methodology use. In addition, there was an additional item included in later agreements for individual assistance to IAEA in the event of a request for assistance in using INTERTRAN from one of the Member States. No requests under this last item were received during the course of the program.

2.0 DEVELOPMENT OF THE INTERTRAN CODE

The INTERTRAN code is an outgrowth of the RADTRAN code developed in the US for the NRC by Sandia National Laboratories [1]. Sandia provided a copy of RADTRAN to Kemakta Konsult, a consultant to Sweden's Nuclear Inspectorate acting for the IAEA, to become the basis of INTERTRAN. Continuing interaction between Sandia, Kemakta and an IAEA Steering Group finally led to the availability of INTERTRAN in 1983 through the IAEA. Sandia has used the code in conjunction with shipment data used in NUREG 0170 for "benchmarking" INTERTRAN results against those in 0170 obtained using an early version of RADTRAN and RADTRAN II results using the same data set. In each comparison case results were found to be comparable, within a few percent, when the actual differences between RADTRAN and INTERTRAN were allowed for (rem/Ci, half lives, photon energies, deposition velocity, accident severity modelling, package dimension factor, lung exposure types, resuspension half-life, ground shine, building dose factor, cleanup level, and uranium cloud area) results agreed closely. Changes made to RADTRAN subsequent to "freezing" of INTERTRAN for publication are not included. This should be evaluated for future changes in INTERTRAN. The changes are discussed in more detail in subsequent sections of this report.

3.0 DEVELOPMENT OF RISK ASSESSMENT METHODS

In the course of RADTRAN use, changes have been incorporated into the code to correct error or omission, and to improve accuracy or remove excessive conservatism as indicated by new data or information. The more interesting of these changes are as follows:

(1) On-link Dose - On major highways with two or more lanes of traffic in one direction, no dose was calculated to people in vehicles alongside the vehicle carrying radioactive material. Since approach distance is smallest of all exposures, adding this dose pathway is significant (20-30% effect on on-link dose).

(2) Rail Dose During Stops - A recent contractor survey indicated time and response distances for persons in rail yards. These data have been included in the stops module of RADTRAN and produce a factor of 2 to 10 decrease in population exposure in rail stops.

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* Research Agreement No. 2793/R1/CF.

(3) Truck Dose During Stops - The study of actual shipment cases indicated that more persons were exposed than originally thought and at somewhat closer distances. The change increased truck dose over earlier versions of RADTRAN. This is a 20-40% effect.

(4) Ingestion Dose - RADTRAN did not include ingestion dose because it was thought to be insignificant. In response to questions on the impact of food-chain doses on results, a simplified and conservative model has been included which uses empirical relationships. The inclusion of ingestion is now fully operational but the resulting impact is insignificant for most nuclides.

(5) Multiple Packages - Large packages or multiple packages that appear to be volume or line sources were not well approximated by point sources. Options were included to allow different specifications and limit doses external to a vehicle to those consistent with regulations.

(6) Accident Risk Spectrum - The use of a single number to characterize accident risk does not adequately portray the relative probability-consequence behaviour of the whole accident spectrum. To provide this kind of information, RADTRAN was modified to accumulate accident consequences and their probability in a rank ordered list by consequence. This list is then processed to provide a plot of probability of exceeding a given level of consequence (in person-rem). This figure is much more useful in understanding and interpreting risk results to interested persons.

In addition to these changes in methodology within RADTRAN, there have been developments in the use of the code which are useful in making risk assessments. Two of specific note are the use of unit risk factors and personal computer pre-processors for RADTRAN.

Unit risk factors are an outgrowth of the observation that for normal exposure situations and for many accident situations the results from RADTRAN evaluation of risk from one type of shipment are linear in distance travelled in a given population density zone and total shipments. Thus it is possible to define a "unit risk factor" (URF) which is specific to a population density zone and has units of person-rem/km/shipment. With URF's (either normal or accident) for a specific type of shipment in specific population zones it is possible to calculate total risk quickly for various shipment scenarios by using the formula:

$$\text{Risk} = \text{NSPTS} * [D_{urban} * URF_{urban} + D_{subu} * URF_{subu} + D_{rural} * URF_{rural}]$$

or

$$\text{Risk} = \text{NSPTS} * \text{DIST} * [f_a * URF_a + f_s * URF_s + f_r * URF_r]$$

where

NSPTS	= No. of shipments
D_u, D_s, D_r	= distance travelled in urban, suburban or rural zones
DIST	= total distance
f_u, f_s, f_r	= D_u/DIST, D_s/DIST, D_r/DIST

The URF's are obtained by calculating risk for 1 kilometre in each zone or by calculating one scenario and taking the output information for one calculation and dividing by the number of shipment-kilometres it includes (taking note of accident/normal and population density zones).

Thus, for application where only an approximate answer is needed or where a number of similar estimates must be done, the unit risk factor may be a reasonable and time effective alternative to full RADTRAN or INTERTRAN evaluations. Some sample URF's are given in reference [2].

Performing these linear combination calculations can be made even simpler by the use of numeric routines that are easily accommodated in something as elementary as a PC "spreadsheet" routine. Sandia has such a routine called RALCOM (RADTRAN combinatorial Algorithm).

Use of RADTRAN and INTERTRAN may also be enhanced by use of file formation routines using PC or other real time input devices. Instead of entering data into the codes directly, input files are formulated and are subject to evaluation algorithms to check that the input data is all there and in the correct range. After such quality assurance techniques have been completed the data files are "swallowed" by RADTRAN to complete the calculation. Our routine to do this is called RADPAC (RADTRAN PAC-man).

4.0 METHODOLOGY USE

The RADTRAN code has enjoyed relatively frequent use for support of DOE waste management programmes and by other organizations for a variety of other programmes. Of particular interest has been the use of RADTRAN to evaluate the risks of transporting spent fuel from reactors to repositories and from reactors to monitored retrievable storage facilities and thence to repositories as required in the US Nuclear Waste Policy Act (NWPA) [3,4]. These documents provide estimates of risks from transport in several categories: radiological (normal), occupational and general public, radiological (accident), non-radiological (normal), general public, and non-radiological (accident) occupational and general public.

In addition to the use of RADTRAN, there have been continuing efforts to obtain better data for support of risk calculations. Of particular note are efforts to more fully define accident risk situations in the water modes and to define normal exposure situations in truck and rail modes. In each case reports will be forthcoming to provide additional support for the best possible modelling of these modes/situations.

There is, of course, continuing data needs throughout the risk models, but one area of particular need is that of package release relationships to accident environments. Work will be started to obtain failure threshold data and correlate it in a way to remove some of the extreme conservatism currently in this section of the risk modelling.

One issue frequently questioned in risk assessment has been sensitivity of the results to variation in parameters. While RADTRAN does have a sensitivity matrix for normal risk, a similar result is not possible for accident risk because of non-linearities associated with threshold events. As a result, a sensitivity evaluation was performed using Monte Carlo methods in order to determine sensitivity of results to random variations in input parameters of RADTRAN. This evaluation is contained in reference 5 and relates to spent fuel transport as described in references 4 and 3.

5.0 CONCLUSION

The US portion of the Agency's CRP has included a broad range of activities from the support of INTERTRAN development to general risk assessment methods development and their use. While it is gratifying to see these methods in use by a wide range of Member States, there is a continuing need for the IAEA to maintain INTERTRAN as a viable tool by assuring rapid correction of errors and prompt dissemination of code enhancements. In that way our efforts and those of other CRP participants in this area will continue to provide mutual benefits.

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ASSESSMENT OF INDIVIDUAL AND COLLECTIVE RADIATION DOSES TO TRANSPORT WORKERS AND POPULATION FROM THE TRANSPORT OF RADIOACTIVE MATERIALS*

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1.0 INTRODUCTION

Problems connected with the transport of radioactive materials are of particular importance in Italy, due to the special attention and awareness of public opinion of the risk of such operations.

Evaluation of individual and collective doses have been carried out in the past years for the transport of spent fuel [1] and for the transport of radioisotopes for medical uses [2]. The computer code INTERTRAN [3], distributed by IAEA for this purpose, was applied with some modifications to evaluate the risk associated with the transport of radioactive materials under normal (incident-free) conditions.

Furthermore, the problem connected with the disposal of radioactive wastes has become a key problem for the development of nuclear energy in Italy, as well as in other countries.

ENEA-DISP, the Italian regulatory authority, is now facing the issue of land disposal of low- and intermediate-level wastes temporarily stored at the production sites. The characteristics of the wastes allow shallow land burial and various areas are suitable to be considered as possible disposal sites. The choice of disposing of the wastes in a single site or in more than one, depends on various factors, some of them quantifiable, others not.

The evaluation of radiation dose to which the general public might be exposed as a consequence of transport of wastes, is an important parameter to be considered in order to discriminate among different options. The present paper deals with the work carried out with the code INTERTRAN to perform such an evaluation.

2.0 COLLECTION OF DATA

The first step in the analysis was the collection of data concerning the wastes. No final disposal sites have been selected in Italy so far and currently the wastes are temporarily stored at the place of production.

It was therefore decided to perform a census of the wastes existing in all nuclear plants, to assess the quantities of materials involved, their origin, physical form and flammability characteristics. The inquiry was not extended to the research centres and hospitals. The wastes produced there are mainly contaminated with short half-live products which can be disposed of as conventional wastes after a suitable period of time.

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* Research Agreement No. 2837/R2/CF.

A list of the nuclear power plants and their geographical location is shown in Table 1.

For practical purposes, wastes are divided by the plant operators into two fundamental types: solids and liquids. Within each type there exist three major categories according to the activity concentration (Table 2). A form was therefore set up containing the information deemed necessary for the evaluation of the radiation exposures of the individuals involved as well as to estimate the quantities of wastes already existing and the future trend of waste production.

Within the form solid wastes are divided in two main groups: combustible and compactable and incombustible and compactable. Within these two groups various types of solid wastes according to their origin have been considered: solid waste, slugs, slurries, ion-exchange resins, ventilation filters and large dimension solid waste.

In Table 3 and 4 the total amount of solid and liquid wastes existing in Italy is reported.

3.0 CHARACTERISTICS OF THE WASTES TO BE TRANSPORTED

The strategy suggested by ENEA-DISP is to divide the wastes according to the half-lives of the radionuclides contained [4].

1st category wastes are wastes mainly originating from medical and industrial applications where half-lives of radionuclides are usually less than two months. They are temporarily stored and after a suitable period are disposed of as conventional wastes.

2nd category wastes are wastes originating mainly from nuclear power plants and contain radionuclides of medium half-lives which can be disposed of in shallow ground.

3rd category wastes are α-bearing wastes and high level β-γ emitting wastes originating from the reprocessing of nuclear fuel. The present analysis is applied to the "2nd category wastes" as they constitute almost 70% of the total amount of wastes produced in Italy.

These wastes, except dry wastes such as rugs, papers, clothes, etc., have to comply with given characteristics. Usually the final products appear in the form of strong industrial packages (220 l drums) in which wastes are compressed or conditioned. Cement is the recommended solidification agent, but ureaformaldehyde resin was also used in the past.

4.0 SHIPMENT SCENARIOS

As it has been pointed out, the characteristics of the site for disposal are not very strict; flat, geologically stable, free from erosion, etc., and different areas are suitable to be chosen for disposal sites.

In the present study four hypothetical sites have been considered, located as shown in Fig. 1 in Northern Italy, in the centre and in the south. In each location there exists a nuclear plant so that the wastes produced there are obviously not moved.

Two different transport modes have been considered: truck mode only and rail plus truck, rail being the primary mode.

The experience gained during the past years when the shipment of irradiated fuel was studied, showed that for each transport mode, owing to the geographical features of the Italian territory, all long distance shipments have almost the same characteristics. This means that for one mode of transport the input parameters required by the code (i.e., fraction of travel in different population density zones, shipment velocity, number of vehicles encountered by the shipment on the way (traffic count), number of shipment stops during the journey, numbers of persons at the stops, etc.) remain the same, as well as the parameters connected with the handling of the packages: persons per handling, handling time, distance handler-package, etc.

In each shipment 50 drums are transported and the average transport index assigned to each package is 0.1 for low activity drums and 0.5 for medium activity drums. An experimental check of these values was carried out during a series of shipments. In Table 5 the values of the input parameters used for each transport mode are reported.

At present, attention is focussed on the normal transport situation and the wastes accumulated at the plants are supposed to be transported in one year.

The radiation exposure of the different population groups considered in the code for each different scenario is reported in Table 6 (truck mode) and Table 7 (rail and truck).

Figure 2 permits a direct comparison between truck mode and rail + truck mode, both for low- and intermediate-activity wastes for each scenario. It appears that collective doses (person-rem) involved in the movement of all the wastes to their final destination are rather low and do not depend strongly on the transport mode. Handlers and crew represent the most exposed group as it was expected, while dose contribution to the population is negligible.

It is important to note that taking into account the trend of the waste production, in regime conditions the exposure of the individuals involved in the movement could be estimated around 1/4 of the values appearing in the tables.

5.0 CONCLUSIONS

This study represents an exercise of the application of the INTERTRAN code to the transport of radioactive wastes under normal conditions of transport.

It must be emphasized that the first and most heavy part of this work has been the collection of data which was very laborious and time consuming. The application of the INTERTRAN code was, then, quite straightforward as the major input parameters were known through previous applications of the code.

Although it is recognized that refinements are needed in the evaluation of the doses to handlers and crew where the point source approximation could not be realistic and in this sense our future work is also oriented [5], it seems that the code gives a means to select among different options.

From the data obtained, it appeared that the doses involved in the transportation, whatever the option, are rather low.

An important point arising from the analysis concerns the groups of most exposed individuals. It appears from the data in Table 6 and 7 that the handlers and crew are such groups. Although the exposures involved are very low (no more than 40 person-rem collective dose, in the worst case) it appears however that it would be advisable to examine the possibility to reduce such exposures by using other types of handling equipment in order to increase distances. The opportunity of using larger packagings in order to reduce the number of handlings should also be examined.

Another point to be emphasized is that, at least in our situation and in normal conditions of transport, there is practically no difference between truck and rail transport mode as far as radiation exposure is concerned.

The situation could change in the case of an accident as accident dynamics and accident rates strongly depend on the mode of transport; it might happen that in this case one transport mode could be preferred to another.

It is therefore deemed very important to give great attention to accident analysis, taking into account also the fact that there exists a category of flammable waste and future studies are oriented in this direction.

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Table 1 - Nuclear Installations in Italy (only those considered in the simulation)

Type of plants	Location
Experimental Reactor (water moderated and graphite refl.)	C.A.M.E.N. - S. Piero a Grado (Pisa)
Experimental Reactor (heavy water moderated and cooled)	C.C.R. - Ispra (Varese)
Experimental Reactors (pool type and sodium cooled)	C.R.E. - Casaccia (Roma)
Reprocessing Plant	EUREX -C.R.E. Saluggia (Vercelli)
" "	ITREC - C.R.E. Trisaia (Matera)
Boiling Water Reactor	Caorso - (Piacenza)
Gas-cooled Reactor	Latina
Pressurized Water Reactor	Trino Vercellese (Vercelli)

Table 2 - Solid and liquid waste classification

Category	Solid	Liquid
High	> 20.0 Ci/m^3	> 10^4 Ci/m^3
Intermediate	0.25 - 20.0 Ci/m^3	0.1 - 10^4 Ci/m^3
Low	0.0 - 0.25 Ci/m^3	10^{-6} - 0.1 Ci/m^3

Table 3 - Summary of solid radioactive wastes in Italy

Plant	Category	Solids			
		not-cond.		cond.	
		m^3	Drums	m^3	Drums
Latina	H	0	0	0	10
	M	72	0	145	14
	L	112	1081	0	0
Trino	H	3	0	0	0
	M	0	0	5	283
	L	0	3059	48	107
Caorso	H	0	0	0	0
	M	0	0	0	158
	L	0	1612	0	6892
Eurex	H	0	101	207	0
	M	0	0	0	0
	L	0	1849	0	0
Itrec	H	0	12	24	0
	M	0	60	0	0
	L	1140	1043	1	0
C.A.M.E.N.	H	0	0	75	0
	M	0	0	0	0
	L	0	0	875	0
Casaccia	H	92	1875	0	0
	M	0	0	0	0
	L	393	5702	104	0
Ispra	H	0	0	427	0
	M	0	0	16	0
	L	0	0	40	6000

Table 4 - Summary of liquid radioactive wastes in Italy

Plant	Category	Liquids	
		quantity m^3	specific activity Ci/m^3
	H	0.0	0.0
Latina	M	0.0	0.0
	L	0.0	0.0
	H	0.0	0.0
Trino	M	0.0	0.0
	L	0.0	0.0
	H	0.0	0.0
Caorso	M	0.0	0.0
	L	0.0	0.0
	H	102.0	4000.0
Eurex	M	95.0	0.7
	L	29.0	0.01
	H	2.7	0.0
Itrec	M	51.0	0.0
	L	0.9	0.0
	H	0.0	0.0
C.A.M.E.N.	M	0.0	0.0
	L	0.0	0.0
	H	8.0	0.001
Casaccia	M	0.0	0.0
	L	102.0	0.0001
	H	0.0	0.0
Ispra	M	0.0	0.0
	L	0.0	0.0

Table 5 - Parameters used for the simulation

Parameter	Unit	Truck Mode	Rail Mode	Truck Mode (secondary)
Population Density				
rural	person/km^2	25	25	25
suburban	"	110	110	110
urban	"	500	500	500
Fraction of travel				
rural	%	0.35	0.70	0.00
suburban	"	0.50	0.20	0.80
urban	"	0.15	0.10	0.20
Shipment velocity				
rural	km/h	60.0	90.0	60.0
suburban	"	45.0	70.0	45.0
urban	"	30.0	50.0	30.0
Traffic count				
rural	Vehicle/h	250	2	250
suburban	"	700	5	700
urban	"	1200	5	1200
Crewmen		2	2	2
Dist. source-crew	m	4.0	150.0	4.0
Handlings		2	2	2
Stop-time (24h)	h	4.0	8.0	4.0
- Pers. exposed		20	50	20
- Average distance	m	20.0	20.0	20.0
Storage time	h	0.0	24.0	0.0
- Pers. exposed		0	100	0
- Average distance	m	0.0	100.0	0.0
Persons per vehic.		2	300	2

Table 6 - Collective doses (person-rem) for various groups (truck mode)

	Low Activity Wastes			
	Eurex	Camen	Casaccia	Itrec
Crew	4.3	4.4	6.7	12.0
Handlers	5.4	5.0	4.6	5.5
Pop. on Link	5.0	5.1	7.7	14.0
Pop. off Link	0.07	0.07	0.1	0.2
Pop. on Stops	0.32	0.29	0.45	0.82
Total	15.0	15.0	20.0	33.0

	Medium Activity Wastes			
	Eurex	Camen	Casaccia	Itrec
Crew	1.2	1.0	0.88	1.7
Handlers	1.3	1.3	1.3	1.2
Pop. on Link	1.5	1.3	1.1	2.1
Pop. off Link	0.02	0.02	0.02	0.03
Pop. on Stops	0.09	0.07	0.07	0.12
Total	4.0	3.7	3.4	5.2

Table 7 - Collective doses (person-rem) for various groups (rail + truck mode)

	Low Activity Wastes			
	Eurex	Camen	Casaccia	Itrec
Crew	0.34	0.29	0.29	0.35
Handlers	11.0	9.9	9.3	11.0
Pop. on Link	0.79	0.71	0.87	1.3
Pop. off Link	0.02	0.02	0.03	0.04
Pop. on Stops	1.0	0.96	1.4	2.5
Storage	1.3	1.2	1.1	1.3
Total	14.0	13.0	13.0	17.0

	Medium Activity Wastes			
	Eurex	Camen	Casaccia	Itrec
Crew	0.06	0.06	0.06	0.06
Handlers	2.6	2.6	2.6	2.4
Pop. on Link	0.09	0.08	0.08	0.10
Pop. off Link	0.002	0.002	0.002	0.003
Pop. on Stops	0.14	0.12	0.11	0.19
Storage	0.31	0.31	0.31	0.29
Total	3.2	3.2	3.2	3.1

Fig. 1 - Hypothetical sites for wastes disposal

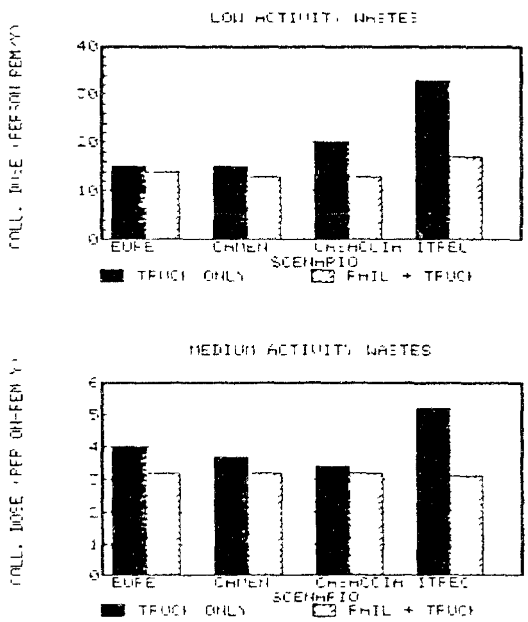

Figure 2 - Comparison between the two transport modes

EVALUATING RISK IN TRANSPORT OF TRITIATED WATER*

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1.0 INTRODUCTION

This report responds to IAEA's Research Agreement 3073/R1/CF on evaluating risk in transport of tritiated water and summarizes a quantitative acceptation criterion based on the limitation of individual risk.

Full load transport situations have been taken into account, while accidental spill cases occurring indoors or in transitory storages were disregarded.

The fundamentals of the acceptation criterion applied in Argentina for evaluating accidental situations in nuclear power plants (NPPs) have been extended to the cases involving the transport of tritiated water. The resulting doses per unit of tritium concentration are evaluated for different accidental situations and a hypothetical example of application of the acceptation criterion is developed.

2.0 INDIVIDUAL RISK

The word "risk" has been used with different meanings in safety evaluations. Individual risk (IR) is herewith understood as the annual probability for an individual to die or suffer severe damage due to potential transport accidents.

IR may be expressed mathematically as follows:

$$IR = P_I \, P_H \, P_D$$

where:

- P_I is the annual probability for a given individual to be involved in a transport accident;
- P_H is the conditional probability for an individual involved in a transport accident to incur a dose H; and
- P_D is the probability for such individual to die or suffer severe damage as a result of dose H.

3.0 QUANTITATIVE CRITERION FOR RISK EVALUATION

The Argentine authority has established a probabilistic criterion in the safety analysis of NPPs based on the maximum individual risk of the members of the public [1,2]. A criterion was adopted for potential accidental situations, by which the individual risk of the most exposed individuals is in the same order of magnitude as that accepted for normal operation.

* Research Agreement No. 3073/R1/CF.

The ICRP [3,4] has recommended that the dose in the critical group should not be above 1 mSv/a, taking into account all radiation sources, except those due to natural radiation and to medical use. Considering this limitation, an individual dose ten times lower - that is 0.1 mSv/a - was considered reasonable as the maximum individual dose due to the normal operation of a single facility. Since the risk derived from exposure to radiation is in the order of 10^{-2} Sv^{-1}, a dose of 0.1 mSv/a implies a risk of death in the order of 10^{-6}/a.

In accidental situations, the individual risk (calculated as the product between the probability of incurring a dose H and the probability of death due to dose H, which was limited to 10^{-6} a^{-1}) implies, at least, double stochasticity.

In practice, no more than ten relevant accidental sequences are observed when performing risk evaluation in NPPs. Therefore, an annual risk of 10^{-7} is considered as acceptable for each one of the sequences. The adopted criterion curve relates the annual probability of occurrence of each sequence with the resulting individual dose for a probability of death of 10^{-7} a^{-1}.

For the dose range in which only stochastic effects may occur, the criterion curve shows a negative and constant slope, represented in a log-log graph of the annual probability of occurrence for the individual dose vs the individual dose itself. The slope ensures that the probability of incurring the dose times the probability of death due to a given dose (considered in the order of 10^{-2} per Sievert) is constant and equal to 10^{-7} per annum. Within the dose range in which non-stochastic effects may occur (e.g., individual doses above 1 Sievert), the curve slope becomes more steep showing a greater risk of death at those dose levels. For doses above 6 Sievert, the probability of death approaches unity and, therefore, the curve remains constant at an annual probability of 10^{-7}. Finally, the curve is truncated by an annual probability of 10^{-2}, since the occurrence of incidents in NPPs involving doses to the public with a higher annual probability is considered as unacceptable. Figure 1 shows the criterion curve used for NPPs.

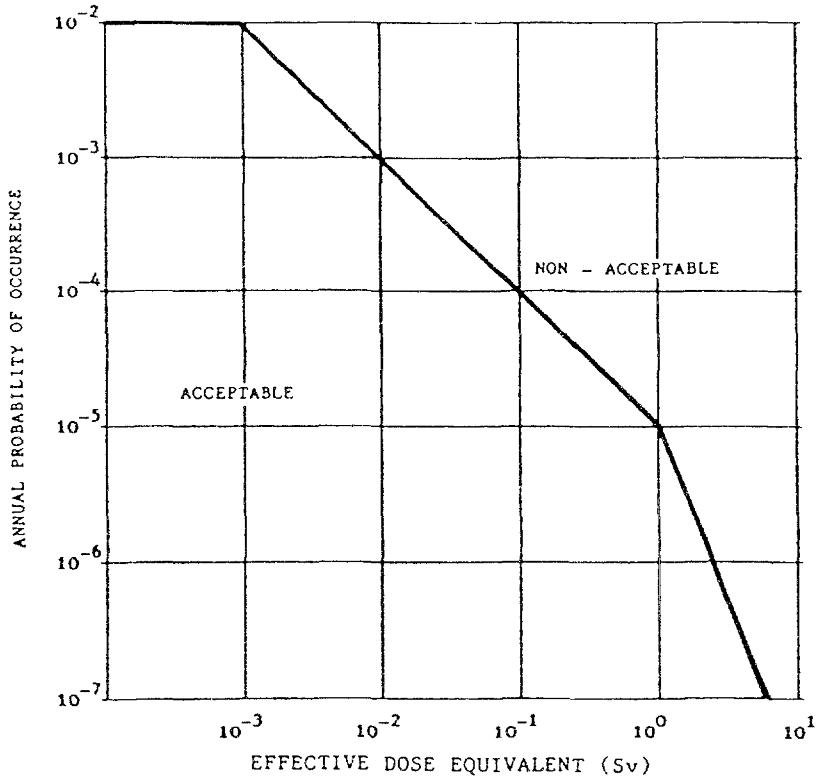

FIGURE 1. CRITERION CURVE

4.0 RISK ANALYSIS FOR A PROGRAMME INVOLVING THE TRANSPORT OF TRITIATED WATER

The individual risk of a programme involving the transport of tritiated water will depend on the probability for an individual to be involved in a transport accident and on the conditional probability for an individual involved in a transport accident to incur a given dose [5,6].

The probability for an individual to be involved in the accident is a function, among other, of the accident rate in the area under consideration and of the transport periodicity. In turn, the conditional probability for an individual involved in the accident to receive a given dose will depend on the exposure model being considered, on the probability of occurrence of a spill with certain characteristics and on the probability of occurrence of given meteorological condition [7,8,9,10,11].

The model used for evaluating the doses that would be incurred by the most exposed individuals in the case of an accident is described in the Appendix. The analyzed accidents were classified by their severity in Categories I through VIII and only those with severity above II will produce doses in the critical group.

The doses resulting from accidents with severity between categories III and VIII, whose relative frequencies range between $7\ 10^{-2}$ and $2\ 10^{-5}$, are only modified in a factor of 2. Besides, the most unfavourable meteorological condition during normal transport operations in Argentina (high evaporation rate) would show an annual frequency above 50%. Therefore, and for practical reasons, the dosimetric factor applied for the model used was 2 mSv Ci^{-1} 1, corresponding with the most unfavourable situation (accident severity VIII and high evaporation rate).

If the maximum concentration of tritium in water in PHWR reactors, in equilibrium, is considered to be lower than 40 Ci/1, the doses to be incurred as per the exposure model under study would fall within the range in which only stochastic effects may be expected.

The acceptation criterion adopted for NPPs may be extended to the transport of radioactive materials if a probability of death of $10^{-6}\ a^{-1}$ is accepted for a whole programme of transport of tritiated water. Thus, an acceptance curve may be drawn as shown in Figure 2, relating the probability of occurrence of accidents in transport with the concentration of the transported tritiated water.

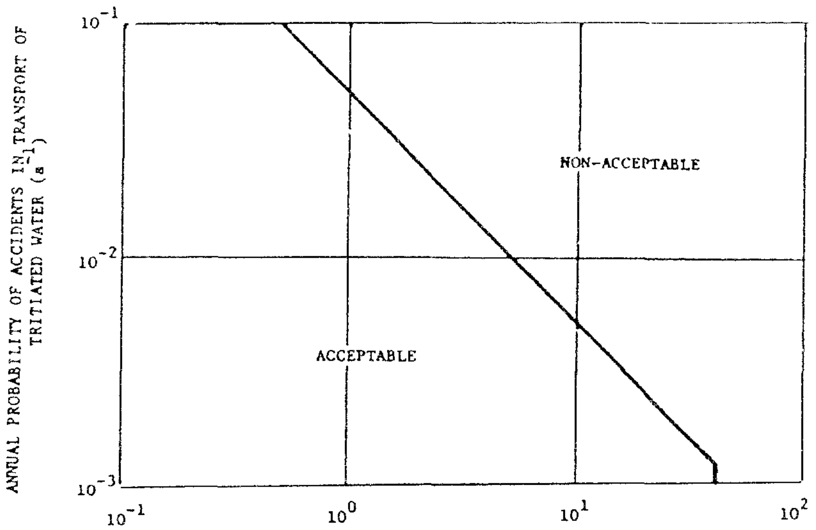

FIGURE 2 - ACCEPTANCE CURVE IN TRANSPORT OF TRITIATED WATER

5.0 EXAMPLE OF APPLICATION

An analysis is made of the acceptability of a programme for the transport of tritiated water in 200 l drums with a concentration of 30 Ci/l. The transport frequency is 120/year. The accident occurrence rate for a given area in the itinerary is estimated in 2 10^{-4} a^{-1} $trip^{-1}$.

The annual probability of occurrence of accidents involving releases of tritiated water is:

$$P_I = 2\ 10^{-4}\ a^{-1}\ trip^{-1}.120\ trip.Fs$$

where:

Fs is the sum of the relative fractions of occurrence of accidents with severity III through VIII that, for the model used, is 0.09 (Table I in the Appendix).

Figure 3 shows the probability of occurrence of accidents, for the example under analysis, as a function of the concentration of the transported tritiated water. It may be observed that the case under analysis is located in the graph within the non-acceptable area and, therefore, either an alternative itinerary will have to be searched in order to reduce the probability of an accident or the tritium concentration will have to be limited to 22 Ci/l.

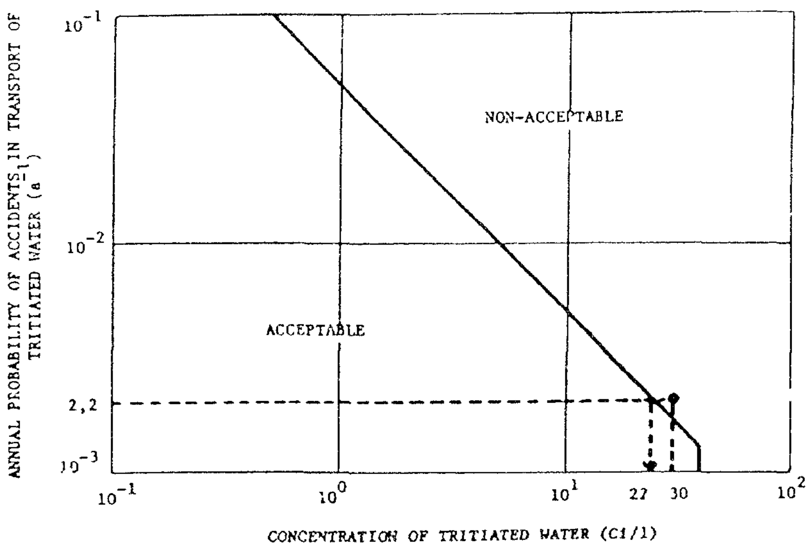

FIGURE 3 - EXAMPLE OF APPLICATION

From a regulatory viewpoint, intrinsic safety (e.g. limitation of tritiated water concentration) is always preferable to limitations imposed to transport operations. Nevertheless, it is necessary to take into account that limitation on concentration could increase the number of transports and consequently the probability of occurrence of accidents.

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APPENDIX

EXPOSURE MODEL

Consideration was given to an accident occurred during the transport of tritiated water and caused by an impact of the transporting vehicle. The case of a fire was not taken into account because, although it would increase the evaporation rate, it would also improve the dispersion conditions and, therefore, it would reduce the resulting individual dose.

It is assumed that there is always an individual involved in a given accident. The dose incurred will depend on the dimension of the spill, on the dispersion conditions of the tritium concentration in the transported water and on the time during which the individual remains exposed in the spill area.

DIMENSIONS OF THE SPILL

It is considered that the water spills on a flat surface and that thickness of the liquid layer is 2 mm. It is also assumed that spills below 200 l will wet a circular area without reaching the road boundaries. The spilled area, A, may be estimated by the expression A = V/e, where V is the spilled volume and e is the thickness of the liquid layer.

For volumes above 200 l, in non-curbed roads, the shape of the water surface will not be circular, since the liquid trespassing the boundaries of the road will be absorbed by the soil. If the spill occurs on a curbed road, the fraction of liquid reaching the curb will flow along the latter and, thus, the spill area will not be significantly increased. Therefore, for the purposes of this evaluation, the size of the water surface resulting from a spill will be independent from the presence or absence of curbs. The effective area, A, expressed in square metres, may be calculated through the expression $A = 8\ V^{1/2}$, where V is the amount of litres spilt [11].

In this case, it was assumed that each vehicle carries 25 drums containing 200 l each. Eight accident categories were analyzed and the spilt volume was estimated by multiplying the transported volume by the fraction corresponding to each category [11]. Table 1 shows the spilt fractions corresponding to each accident severity under analysis.

DISPERSION CONDITIONS

Once the spill has occurred, the tritium dispersion in air will depend on the evaporation rate of the spilt water. The evaporation rate per surface unit, r, has been calculated by applying the following expression:

$$r = 20\ \{\ 0.00274\ u^{0,78}\ \frac{(Ps-Pa)}{R\ Tp}\ \}$$

where:

- r is the evaporation rate per surface unit ($g\ m^{-2}\ s^{-1}$);
- u is the wind speed (m/s);
- R is the universal constant of gases: $8.205\ 10^{-5}\ m^3$ atm/°K mol;
- T_p is the temperature of the spilt water (°K);
- Ps is the pressure of the D_2O saturated vapour at the spill area at the temperature of the spilt liquid (atm); and,
- Pa is the pressure of the H_2O saturated vapour at room temperature (Ta) and relative humidity (H), (atm).

TRITIUM CONCENTRATION

The concentration of tritiated water vapour in the air is calculated by means of the following expression [11]:

$$C = \frac{0.89\ r}{u} A^{1/2}$$

where:

A is the spill area (m^2);
r is the evaporation rate per surface area ($g\ m^{-2}\ s^{-1}$); and
u is the wind speed (m/s).

The values for tritium concentration in air were estimated by considering high, medium and low evaporation rates. The parameters corresponding to each one of these dispersion conditions are summarized in Table 2.

EFFECTIVE EQUIVALENT DOSE

It was assumed that the individual involved in the accident remains immersed in the cloud during 30 minutes. Also, it was taken into account that tritium incorporation occurs through inhalation and through the skin and that both means contribute to irradiation in a similar way [13]. A respiratory flow of 20 m^3/d and a dosimetric factor of 0.84 Sv/Ci were assumed [14].

Table 3 shows the values calcuated for the effective equivalent dose per unit of tritium concentration in the transported water and for different accidental conditions with high, medium and low evaporation rates.

TABLE 1
FREQUENCY IN THE OCCURRENCE OF ACCIDENTS AND FRACTION OF THE RELEASED LOAD [8]

Degree of Severity	Relative frequency of occurrence (Fs)	Released fraction
I	0.55	0.00
II	0.36	0.00
III	0.07	0.05
IV	0.016	0.1
V	0.0028	0.2
VI	0.0011	0.5
VII	$8.5\ 10^{-5}$	0.9
VIII	$1.5\ 10^{-5}$	1.0

TABLE 2

DISPERSION PARAMETERS

Evaporation rate	Room Temperature °C	Spill Temperature °C	Relative Humidity %	Wind Speed m/s
High	35	50	15	10
Medium	16	20	72	3.5
Low	5	5	100	0.5

TABLE 3

DOSES PRODUCED PER UNIT OF TRITIUM CONCENTRATION
AS PER THE SEVERITY OF THE ACCIDENT AND THE DISPERSION CONDTIONS

Degree of Severity	EFFECTIVE DOSE EQUIVALENT (m Sv Ci^{-1} . 1)		
	High	Medium	Low
I	0	0	0
II	0	0	0
III	0.90	0.15	0.03
IV	1.06	0.17	0.03
V	1.27	0.21	0.03
VI	1.60	0.27	0.04
VII	1.83	0.30	0.05
VIII	1.90	0.31	0.05

CONTRIBUTION TO POPULATION DOSE FROM THE TRANSPORT OF RADIOACTIVE MATERIALS IN THE PHILIPPINES*

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1.0 INTRODUCTION

The purpose of this study is to assess the radiological impact to the population from the transportation of radioactive materials in the Philippines. Actual dose measurements are being made, extrapolations to the population of the country will be performed in part using the INTERTRAN computer code.

This study was initiated late in the Coordinated Research Programme, and therefore the following is only a brief interim report.

2.0 SUMMARY OF ACCOMPLISHMENTS

The scientific scope of the project involves the following: (a) determine the radiation doses to the population from the transport of radioactive materials in the country, (b) develop an empirical relation between actual doses received by transport workers and handling distances, (c) correlate measured doses with reported transport indices, and (d) evaluate results in terms of the INTERTRAN code.

During the first year of the study, statistical data relevant to the transport of radioactive material in the country were collected. This includes other parameters necessary for use in the INTERTRAN calculations. Shipment data covering the year 1983 and 1984 are shown in the attached Tables.

One significant activity undertaken during the first year of the study was the adaptation of the INTERTRAN code on our VAX-11/750 computer system with very few modifications. Some corrections which were required to the INTERTRAN code were made on our VAX version without any difficulty. The code was successfully run using test data producing the desired output.

For the second year of the study, actual dose measurements of transport workers during normal transport conditions were initiated using thermoluminescent dosemeters and pocket ionization chamber dosemeters. Actual handling distances, average time spent by the individual in the vicinity of the packages, number of packages handled, surface dose rates, transport indices and distances travelled were among the parameters taken into consideration. A computer program is being developed for extrapolating data from these measurements to the whole population in the country.

Preliminary results on dose measurements have shown reasonable agreement between calculated and measured values.

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* Research Agreement No. 3744/R1/RB.

Due to constraints in time and sampling resources as well as problems related to documentation and record management, a lot of effort has to be spent in the collection of shipment data. This is clearly necessary before the INTERTRAN code is used to ensure validity of results. It is therefore proposed to undertake activities relevant to the collection of shipment data to complete this as accurately as possible in the light of unreported shipments/non compliance to the regulations and the like.

Table I

RADIOACTIVE MATERIALS TRANSPORTED IN THE PHILIPPINES
BY ROAD, SEA & AIR (1983 & 1984), BY PACKAGE TYPE

MODE / PACKAGE CLASSIFICATION	NUMBER OF PACKAGES 1983	NUMBER OF PACKAGES 1984	TI 1983	TI 1984	DISTANCE (km) 1983	DISTANCE (km) 1984
I. ROAD						
EXEMPT	1,172	1,365	-	-	10,760	9,840
LSA or LLS (not full load)	-	-	-	-	-	-
Type A	1,090	1,739	0 - 4	0 - 4	13,141	18,129
Type B(U)	26	15	0 - 7	0 - 7	2,306	4,445
Type B(M)	-	-	-	-	-	-
Special Arrangement	-	61	-	0 - 1.2	-	400
Full Load	-	-	-	-	-	-
II. SEA						
EXEMPT	-	-	-	-	-	-
LSA or LLS (not full load)	-	-	-	-	-	-
Type A	2	-	0 - 4	-	2,800	-
Type B(U)	8	1	0 - 7	0 - 7	16,000	2,000
Type B(M)	-	-	-	-	-	-
Special Arrangement	-	-	-	-	-	-
III. AIR						
EXEMPT	48	100	-	-	43,200	90,000
LSA or LLS (not full load)	-	-	-	-	-	-
Type A	-	-	-	-	-	-
Type B(U)	20	4	0 - 7	0 - 7	17,740	8,000
Type B(M)	-	-	-	-	-	-
Special Arrangement	-	-	-	-	-	-
Full Load	-	-	-	-	-	-

Table II
RADIOACTIVE MATERIALS TRANSPORTED IN THE PHILIPPINES
BY ROAD, SEA & AIR (1983), BY TYPE OF CONTENTS

MODE / TYPE OF MATERIAL	NUMBER OF CONSIGNMENTS	NUMBER OF PACKAGES	ACTIVITY (Ci) TOTAL	ACTIVITY (Ci) MAX. PER CONSIGNMENT	TI	DISTANCE (km)
I. ROAD						
Sealed Source (Medical)	1	1	5,288	5,288	0-3	6
Radiopharmaceuticals	300	2,176	70	0.900	0-4	20,889
Industrial Radiography	26	26	1,800	160	0-7	2,300
Other Industrial Sources	10	10	0.5	0.1	0-3	3,000
U & Th Ores & Concentrate	2	75	5000 kilos	2500 kilos	0-2	12
II. SEA						
Sealed Source (Medical)	-	-	-	-	-	-
Radiopharmaceuticals	-	-	-	-	-	-
Industrial Radiography	8	8	560	100	0-7	16,000
Other Industrial Sources	2	2	0.400	0.200	0-4	2,800
U & Th Ores & Concentrate	-	-	-	-	-	-
III. AIR						
Sealed Source	-	-	-	-	-	-
Radiopharmaceuticals	48	48	microcurie level	microcurie level	-	43,200
Industrial Radiography	20	20	1,500	100	0-7	17,740
Other Industrial Sources	-	-	-	-	-	-
U & Th Ores & Concentrate	-	-	-	-	-	-

Table III
RADIOACTIVE MATERIALS TRANSPORTED IN THE PHILIPPINES
BY ROAD, SEA & AIR (1984), BY TYPE OF CONTENTS

MODE / TYPE OF MATERIAL	NUMBER OF CONSIGNMENTS	NUMBER OF PACKAGES	ACTIVITY (Ci) TOTAL	ACTIVITY (Ci) MAX. PER CONSIGNMENT	TI	DISTANCE (km)
I. ROAD						
Sealed Source (Medical)	-	-	-	-	-	-
Radiopharma-ceuticals	169	3,084	41	0.400	0-4	27,933
Industrial Radiography	15	15	-	100	0-7	4,445
Other Industrial Sources	7	7	43.4	50	0-4	24
U & Th Ores & Concentrate	2	13	1000 kilos	600 kilos	0-2	12
UO_2	2	61	1,290,150gm	21,150	0-1.2	400
II. SEA						
Sealed Source (Medical)	1	1	0.400	0.400	0-7	1,000
Radiopharma-ceuticals	-	-	-	-	-	-
Industrial Radiography	1	1	100	100	1-7	2,000
Other Industrial Sources	-	-	-	-	-	-
U & Th Ores & Concentrate	-	-	-	-	-	-
III AIR						
Sealed Source	-	-	-	-	-	-
Radiopharma-ceuticals	75	100	microcurie level	microcurie level	-	90,000
Industrial Radiography	4	4	300	100	0-7	8,000
Other Industrial Sources	-	-	-	-	-	-
U & Th Ores & Concentrate	-	-	-	-	-	-
UO_2	-	-	-	-	-	-

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