**SPECIALISTS' MEETING ON HANDLING
AND TRANSPORTATION OF SPENT
FUEL ELEMENTS FOR LMFBR'S**

Sponsored by the International Atomic Energy Agency
International Working Group on Fast Reactors
Rome, Italy, 26-28 April 1972

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U.S. Atomic Energy Commission)

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FUEL RECYCLE
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Category UC-79g
FUEL HANDLING

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INTERNATIONAL ATOMIC ENERGY AGENCY
International Working Group on Fast Reactors

SPECIALISTS' MEETING ON HANDLING AND TRANSPORTATION
OF SPENT FUEL ELEMENTS FOR LMFBR'S

Rome, 26-28 April 1972

Chairman: K.J. HENRY, United Kingdom
Scientific Secretary: E.I. INVUTIN, IAEA
Secretaries of the Meeting: G. CRISTIANI, L. FAILLA and
A. MARZOCCHI, Italy
Summary of proceedings and conclusions prepared by
K.J. HENRY and G.F. HEAP, United Kingdom

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2. SUMMARY OF PROCEEDINGS AND CONCLUSIONS

1. Present status of the International laws regulating transport of fuel for fast reactors.

1.1. The IAEA regulations issued on 1967.

1.2. The proposed revision of said regulations.

The meeting noted that the committee responsible for the review of the IAEA Regulations for the Safe Transport of Radio-active Materials had recently agreed that the 1967 Regulations, with some minor modifications concerning the leakage criteria, should be adopted in the 1972 edition. It was also noted that the revision is in process of ratification.

1.3. Possible suggestions and proposed amendements.

France suggested that they would prefer to see the containment specified in terms such that they could have the option of adopting either a volume leakage rate or an activity leakage rate.

1.4. Regulations and their impact against possible accidents.

All Countries are at present employing the 1967 IAEA Regulations with regard to test standards. It was stated that in the United Kingdom, the C.E.G.B. and British Railways were discussing the implications of the expected higher transit speeds of both goods and passenger traffic over the next decade on the test requirements, and in consequence were considering the advisability of increasing the specified drop height in view of the accidental impact possibilities. It was also noted that the competent Authority within the United Kingdom were not convinced that a more stringent mechanical test should be adopted.

Transnuclear GmbH reported that they had carried out 20 meter drop tests on to a less rigid target in another context.

They had no intention of modifying the normal test standard for transport purposes.

No other Countries reported any consideration of different impact standards being undertaken in relation to LMFBR fuel.

2. Experience in transport of fuel for fast reactors.

2.1. Transport of non irradiated fuel.

Japan reported experiments on the road transport of LMFBR fuel assemblies which demonstrated only very minor damage. General discussion took place on the problem of ensuring the integrity of fuel assemblies delivered to the reactor, since final inspection normally took place at the fuel fabrication stage. A Japanese suggestion that recording accelerometers should be attached to the delivery vehicles received favourable acceptance.

It was agreed that little technical difficulty exists in the transport of unirradiated fuel.

2.2. Transport of irradiated fuel.

France reported the successful transport of about 10,000 fuel pins from the Rapsodie Reactor to La Hague. These movements had been made by road: the choice of the method of transport was taken on economic grounds. It is intended in the first period of operation to transport Phenix fuel using the same system.

The question of whether it would be necessary to provide special containment for damaged fuel arose in the context of a statement made by Germany F.R.: it had been known for up to 50% of LWR fuel elements to be "leaking" on removal from the reactor. It was generally considered that provided the flask itself met the required containment standards, additional internal containment,

i.e. a sealed canister, would not be essential. It was noted that in the case of LMFBRs, a substantial proportion of "leaking" fuel would cause sodium contamination of reprocessing plants.

The question of "damaged" fuel in which oxide is released from the pin was accepted as a different consideration which could well result in a requirement for separate internal containment.

France and the United Kingdom were both considering the provision of a separate internal containment for LMFBR fuel.

2.3. Transport by truck, train, ship and airplane.

Italy reported that they had successfully transported a 12 ton fuel flask by air to Glasgow. It was generally agreed that there are no particular difficulties attributable to road, air, rail or ship transport other than those arising from the flask weight and external dimensions.

2.4. Possible damage of fuel elements and handling of damaged fuel.

There was no report of any problem of fuel element corrosion arising during transport.

As previously mentioned it was agreed that damaged fuel could be carried with normal containment unless there was a requirement for a larger hole size or short decay time to enable early examination to be carried out. In this case a special single fuel assembly flask would probably be required.

2.5. Decontamination of containers.

No unusual difficulties were reported.

2.6. Accidents: statistics and suggestions.

The participants reported that they had no knowledge of any serious accidents occurring in the transport of irradiated fuel.

3. Containers for transport of fuel for fast reactors.

3.1. Problems of design and construction of the containers presently used for both irradiated and non irradiated fuel.

France, Germany F.R. and the United Kingdom tabled papers on their respective design proposals for FBR fuel flasks. The French and British designs incorporated separate internal containments for assemblies. The German design used helium cooling for SNR fuel assemblies, but it was agreed by all parties that there appeared to be no alternative to the use of liquid coolants within the flasks, for the high heat output fuels of commercial fast reactors. The French and British proposals incorporated the use of sodium as a coolant. The U.K. design was capable of dissipating a total of 40 kwatts decay heat from 5 assemblies.

Transnuclear experimental work indicated that it might be possible to dissipate up to 80 kwatts from a copper-finned sodium filled flask, without forced cooling.

The problem of neutron shielding was discussed and all parties concerned indicated that they were investigating the use of water and boron impregnated plastics as shielding materials. The dependence of the neutron dose rate on the pre-irradiation history of the fuel was noted.

Depleted uranium has not yet been considered for the gamma shielding material although it was recognized that it may assume importance in the future in order to increase the capacity of FBR flasks.

3.2. Handling of containers.

Papers tabled by France and Germany F.R. described the handling arrangements for the Rapsodie and SNR reactors. Both arrangements were essentially similar employing top loading of transport flasks. France

reported similar arrangements for Phenix.

3.3. Standardization of containers.

Until fuel elements, reactor and plant load/off load facilities become more standardized, perhaps in 10 year time, it was not thought possible to promote any substantial degree of flask standardization.

3.4. Future development of containers.

Transnuclear reported having carried out experiments using electrically heated fuel element sections to establish the heat transfer performance in helium, and intend to extend this work to cover sodium coolant. Some investigations of organic and metal alloy coolants will also be carried out.

Similar work has been carried out by France for assemblies in air which it is proposed to extend to sodium.

The U.K. are about to embark on a series of drop tests on full scale containment cannisters and 1/4 scale model flasks. The drop height employed may be greater than the current 9 meter IAEA standard.

Forced cooling of LMFBR flasks was discussed, as a method of increasing the heat loading. It was generally agreed that due to the operational difficulties which would arise, e.g. ensuring maintenance of power supplies, etc., natural cooling systems were preferred. In principle there was no objection to forced cooling, so long as it could be shown that loss of cooling would not result in loss of containment.

5. Economics

The main discussion centred on the problems posed by the possible requirement for the reprocessing of short cooled fuel.

Both France and U.K. considered that 180 day cooled LMFBR fuel could be processed using presently available aqueous processing techniques and that these could probably accommodate 90 day cooled fuel. Work is being undertaken in both Countries to demonstrate this capability. For much shorter cooled fuel (say 60 days) a different processing technology may have to be employed. It was recognized that the cost of processing increased with shorter cooling times and that this extra cost would have to be balanced against savings in fuel inventory. The costs of handling and transport would also increase with shorter cooling times. Details of the extent of processing cost increases were not available. A paper tabled by France indicated a 0.2% reduction in kw^h costs arising from inventory savings if the cooling time were reduced by 30 days from 180 days, assuming fixed transport and reprocessing costs. Transnuclear reported estimates which showed a 20% increase for the transport costs of 40 day cooled fuel in sodium, over 100 day cooled fuel in helium in similar flasks.

No special problem associated with the insurance of LMFBR fuel transport was anticipated. A current cost of 100-200 dollars per movement for a maximum cover of 10 million dollars, was reported from France.

It was noted that the experience to date has not revealed any insuperable problem arising from the shipping of irradiated fuel across international frontiers. It was not expected that LMFBR fuels would present any special problem.

6. Conclusions.

It appeared that the design proposals of the countries involved were following similar lines, i.e. naturally cooled flasks,

liquid sodium cooling, double containment, etc.

The main problem areas are in the design of the flasks and more information is urgently required in the following areas:

a) Fuel cooling periods

What are the economics of decreasing fuel cooling period?

Flask designs cannot properly proceed without this information.

b) Heat transfer

Little experimental data is available on heat transfer and temperature distributions in flasks of this type. More experimental work is required.

c) Neutron emissions

Further evaluation of neutron emission ratio is required.

d) Seals

Sealing arrangements assume ever greater importance in these designs than previously; work is required in the evaluation of seals under the conditions of high temperature, impact and corrosive environments which will exist.

3. ANNEXES

a) LIST OF THE TOPICS OF THE MEETING

1. Present status of the International laws regulating transport of fuel for fast reactors:
 - 1.1. The IAEA regulations issued on 1967.
 - 1.2. The proposed revision of said regulation.
 - 1.3. Possible suggestions and proposed amendements.
 - 1.4. Regulations and their impact against possible accidents.
2. Experience in transport of fuel for fast reactors:
 - 2.1. Transport of non irradiated fuel.
 - 2.2. Transport of irradiated fuel.
 - 2.3. Transport by truck, train, ship and airplane.
 - 2.4. Possible damage of fuel elements and handling of damaged fuel.
 - 2.5. Decontamination of containers.
 - 2.6. Accidents: statistics and suggestions.
3. Containers for transport of fuel for fast reactors:
 - 3.1. Problems of design and construction of the containers presently used for both irradiated and non-irradiated fuel.
 - 3.2. Handling of containers.
 - 3.3. Standardization of containers.
 - 3.4. Future development of containers.
4. Economics
 - 4.1. Containers.
 - 4.2. Handling.
 - 4.3. Actual transport.
 - 4.4. Insurance.
 - 4.5. International problems.

b) TIME-SCHEDULE

26th April 1972

Opening	9.30 a.m.
Coffee break	11.00 a.m.
Lunch	1.00 p.m.
Afternoon session	from 3.00 p.m. to 5.30 p.m.

27th April 1972

Opening	9.00 a.m.
Coffee break	11.00 a.m.
Lunch	1.00 p.m.
Afternoon	2.30 p.m. Visit to Casaccia Nuclear Research Center

28th April 1972

Opening	9.30 a.m.
Coffee break	11.00 a.m.
Closure	12.30 a.m.
Lunch	1.00 p.m.

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d) PAPERS PRESENTED AT THE MEETING

- CHRIST R. - Cask design for spent SNR fuel elements.
- WACNER J. - Handling and loading of irradiated SNR-300 elements.
- Problèmes de conception des emballages servant au transport entre la Centrale et l'Usine - Sommaire d'une étude se rapportant au sujet 3.1.
- AUBERT M. - Aspects des recommandations internationales concernant les transports des combustibles des reacteurs rapides.
- LABROUSSE M. - Experiences francaises dans le transport des combustibles irradiés de réacteurs rapides.
- LENAIL B. - Problèmes posés par la réception des combustibles de réacteurs rapides dans les usines de retraitement.
- UCHIDA I. - Experimental transport of Dummv fuel subassemblies for "Joyo".
- HENRY K.J. - Transportation of LMFBR fuel elements.

INTERNATIONAL ATOMIC ENERGY AGENCY

SPECIALIST MEETING ON HANDLING AND TRANSPORTATION OF
LMFBR SPENT FUEL ELEMENTS

Rome, 26-28 April 1972

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CASK DESIGN FOR SPENT SNR FUEL ELEMENTS

by

R. CHRIST - Transnuclear GmbH, Germany

Cask design for spent SNR fuel elements

1. Introduction

A 300 MW fast breeder demonstration plant will be built in Germany in the near future. The fuel elements will be bundles of 169 rods in hexagonal array surrounded by a hexagonal shroud tube. The objective of a study performed by Transnuklear GmbH was to evaluate the use of LWR fuel cask technology for the transport of spent SNR fuel elements.

2. Cask design (Fig. 1 and 2)

The main features of the cask design are

- heat transfer within the fuel bundle by Helium
- external heat transfer by natural convection
- n-shielding at the surface of the cask
- balsa wood shock absorbers at the extremities of the cask.

These points are explained in detail as follows:

The cask has 6 - 7 lodgements (the central location is intended for canned fuel elements with a rather long cooling time) arranged in hexagonal array. The lodgements are closed individually by lead shielded plugs. The pitch of the array is determined by the way of which the cask is loaded at the reactor site. Heat transfer within the lodgements will be by radiation and conduction in Helium. The aspects of Helium cooling have been evaluated in detail through experiments and calculations. The decay heat is transformed to the outer shell of the cask by conduction through the lead shielding. Axial heat distribution will be improved by introducing a highly conductive material between or close to the fuel element lodgements.

At the surface, the cask consists of long copper blades with excellent heat transfer characteristics. For neutron shielding about 150 mm of hydrogenous material is poured between the fins. As the copper fins are extending through the neutron shield heat transfer is not reduced.

3. Experiments and calculations related to heat transfer

3.1 Heat transfer within the fuel element lodgement

Experiments with a SNR fuel rod bundle were performed

using Helium and other gaseous heat transfer media (Fig. 3). The results are in very good agreement with calculations performed by Interatom (Fig. 4).

3.2 Heat transfer in the shielding

One of the major problems in fast breeder fuel cask design is to distribute the decay heat released in the short active fuel zone to the large surface of the cask given by the total length of the elements. Computer calculations were performed to get information about the temperature distribution along the axis of the cask (Fig. 5).

3.3 Efficiency of copper fins for convective heat transfer

To evaluate the efficiency of the copper fin system a full scale cask surface with reduced length was built (Fig. 6). The mock up was internally heated to different power levels. The results (Fig. 7) show that at a diameter of 1,7 m up to 15 kW/m of cask length can be transmitted without exceeding the 82 °C limit of the regulations (at 38 ° ambient). Even 20 kW/m can be transmitted if a surface temperature of 90 °C is accepted.

4. Temperatures attained in the cask during normal transport conditions

Assuming 6 SNR elements in the cask, the temperatures at 100 d decay time (= 16,5 kW) will be as follows:

Cask surface	82 °C	
Maximum lead temperature	150 °C	
Temperature of the shroud tube:	360 °C	
Central fuel rod	580 °C	(He)
	650 °C	(He + 10 % fission gases)

The design temperature limit for the fuel rods is 700 °C. So there is a safety margin even for the case of Helium contamination by release of fission product gases

5. Performance under accident conditions

The closures of the fuel elements are protected against distortion by balsa wood shock absorbers at the extremities. There also are lateral shock absorbers at top and bottom of the cask to reduce damage to shielding and fins. As there is no forced cooling at the surface of the cask, the 9 m drop will not affect heat transfer except for local damage of fins.

Excessive heat input into the cask during fire is avoided by a layer of wet plaster between lead and outer shell of the cask. The fire will dehydrate the plaster and thus reduce its conductivity by a factor of 10. Also the shock absorbers are of low conductivity. Fuel rod temperature will not increase by more than 10 °C during the fire test.

The behaviour under accident conditions has been tested successfully with cask models.

6. Conclusions, future developments

Use of LWR cask technology for IMFBR fuel shipping offers reliable solutions for the cases where short decay times before transport are not or not yet mandatory.

It is understood however that this is no longer valid for shipments of large amounts of fast breeder fuel elements after short decay times. Sponsored by the German government Transnuklear is therefore preparing experiments to evaluate liquid heat transfer media including sodium, lead alloy and organic coolants and will study steel as an alternative shielding material.

Fig. 1

Cask for SNR
fuel elements

Cask for SNR fuel elements
Cross section
Pg. 21

Fig. 2

Original photograph for
reproduction not available!

Fuel bundle and shroud tube

Electrical heaters

Fig. 3 Heat transfer experiment with a
SNR fuel bundle

Central rod temperature in the SNR fuel rod bundle filled with Helium

Fig. 4

Calculated temperature profile
along the cask at 38°C ambient

Fig. 5

Pg. 24

Fig. 6 Heat transfer experiment with
copper fins

Results of heat transfer experiments with copper fins

Fig. 7

Specialists' Meeting on Handling and Transportation
of LMFBR Spent Fuel Elements

Rome, 26-28 April 1972

Handling and Loading of Irra-
diated SNR-300 Elements

J. Wagner

INTERATOM
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Bensberg / Germany

1. Introduction

Construction of the SNR-300, a 300 MWe Fast Breeder Prototype Reactor, is scheduled to start before the end of this year. It is to be built by a consortium comprising BELGONUCLEAIRE (Belgium), NERATOOM (Netherlands) and INTERATOM (Germany).

Irradiated fuel and blanket elements will be reprocessed probably in the WAK Karlsruhe. For loading and shipping the elements, suitable equipment has been foreseen in the SNR plant. This paper deals with that equipment.

2. Design criteria

After refuelling the irradiated elements are stored in a sodium-filled tank, each element in a separate pot. They remain within the reactor plant for a period of about 100-150 days. The decay heat is then in the order of magnitude of 3 kW; only there after the elements will be accepted by the reprocessing plant.

It is envisaged to ship the elements contained in a helium atmosphere. The decay heat is transferred to the container outside by means of radiation and conduction, regarding which tests and calculations have been performed by INTERATOM and TRANSNUCLEAR. Prior to shipping, the elements are not cleaned. Adhering sodium amounts are small because of the "blowing-down effect" during forced argon cooling in the fuel handling machine. For maximum availability of the reactor plant, loading of the shipping container is done only during reactor operation.

3. Handling and loading of irradiated elements (see Figure)

After cleaning down the shipping container, the truck is driven into the service station within the air-lock area. There the container is prepared for the loading process: the shock absorbers are removed and stored on the truck; the container is raised to a vertical position and reloaded onto a special trolley: the interlocks of the container-plugs are disengaged and the plugs are lifted. By transverse movement of the trolley and rotation of a support-slab on the trolley, all container positions can be positioned under the packing station lower gate valve.

The packing station is provided with a movable shield (it lowers, when the container is positioned, and seals as well as shields the inner parts of both station and container), shielded gate valves, a rotating magazine with a plug hoisting mechanism, and systems for argon, helium and waste gas (vacuum). The station acts as the air-lock between the reactor containment and the control area outside the containment (truck air-lock).

The task of the rotating magazine is to pull the plugs into the magazine, to form a guide-tube for gripper and element, to lower and finally to push the plug back into the container. Before an element is inserted into a container position, the air is replaced by argon using the waste-gas and argon-system. After lowering the element and closing the upper gate valve, but before insertion of the plug, the argon is replaced by helium. The final operation of the packing station is the rough control of gas-tightness of the container position. Each container-position is treated separately.

The fuel handling machine withdraws the irradiated elements from the sodium filled pots in the storage and transports them to the packing station. After the gas-tight connection between machine and station has been made, the gate valves are opened and the element is lowered into the container-position. Further conditions before element lowering are the operations mentioned above, namely plug hoisting and changing the argon atmosphere.

When the container is filled, the special trolley moves back to the service station. Here the following steps are foreseen:

- Adjustment of the proper helium pressure within each container position by connections to the helium and waste-gas systems and the container respectively, in order to take into account the unsteady - state conditions during loading.
- Exact control of gas-tightness of each container-position. For that purpose a measuring head with connections to vacuum and pressure gauges is anticipated.
- Mounting of an additional common sealing-lid for all positions.

To complete the sequence of operations, the container is set back on the truck. The shock absorbers are reinstalled, and shipment to the reprocessing plant can start.

4. Handling of unirradiated fuel

The handling is basically the same as for irradiated elements. Main differences include:

- . The containers are not so heavily shielded
- . No helium is used
- . No cooling is necessary

Equally, new elements are handled by the fuel handling machine because of a substantial neutron radiation dose (20 mrem/h, 30 cm from the surface of a single element; 300 mrem/h in the storage for new elements by $k_{eff} = 0.4$). They are stored in an argon atmosphere. All new elements are checked in a hot cell (observation, measuring of outer dimensions).

MEETING OF SPECIALISTS ON TRANSPORTATION OF SPENT FUELS FROM FAST REACTORS

/Paper; Rome, Specialist Meeting on Handling and Transportation of Spent Fuel Elements for LMFBR, French, 26-28 April 1972, pp 1-12/

Summary of Study Pertaining to Item 3.1

Problems in designing packaging used in shipments between the power station and the fuel reprocessing plant.

1. Purpose

We must first of all estimate how much the kilowatt-hour cost is decreased by an overall reduction in the fuel-cycle period, whatever the cause of that reduction may be. This decrease in kilowatt-hour cost constitutes the study's economic justification.

Within the scope of this study, the reduction in the duration of the fuel cycle may be obtained by:

a much earlier initiation of fuel shipment. This entails the transportation of fuel assemblies having a high residual power, and the reprocessing of these assemblies at a slightly lower power. Any check or limit placed on the power acceptable for reprocessing nullifies the advantage of any shipment at a power greater than that limit.

by reducing transportation time. This requires rapid rotation of shipping casks between reprocessing plant and power stations, a sufficient inventory of casks, a high cask loading and unloading rate, and a reprocessing schedule or rate equivalent to the shipment schedule or rate. Any check or limit on the reprocessing rate can appreciably reduce the advantage of shipment at a greater rate than this check.

2. Scope of the Study

The different problems were studied within the framework of the development of a group of power stations with a unitary rated power output of 1200 megawatts electrical.

The following assumptions were also made:

a. A reduction in the facilities at the power station site and consequently:

shipment of complete fuel assemblies (solely the "foot" may possibly be cut off, but this parameter is not applicable at this stage of the study);

storage capability at the power plant, with a capacity that must not be influenced by the shipment rate;

only a summary cleaning of assemblies by blowing is planned.

b. Simplified design of shipping casks.

To this end we have deliberately excluded any solution whereby the assemblies are cooled by forced circulation of a fluid. We prescribe a temperature limit of 650°C for the cladding of the center pin. It must be pointed out, however, that in the case of forced circulation, the possibility of a stoppage in that circulation should be anticipated. Should such stoppage occur, we would once again find ourselves in the case under discussion with the sole difference being the possibility of accepting a higher cladding temperature (approximately 800°C , and even higher should the cladding be destroyed).

3. Economic Justification

We may analyze to what extent various factors affect fuel-cycle cost, particularly such a factor as the elapsed time between the unloading of the spent fuel assemblies and the conclusion of their reprocessing. Calculations made for elapsed times of from 4 to 15 months, shows that for each month saved on the fuel cycle there is a resultant saving of 1.25 percent in the cycle cost. With the estimated cycle cost of 0.41 centime/KWH and based on a total price of 2.5 centimes/KWH, we see that each month saved on the fuel cycle results in a saving of approximately 0.2 percent in the cost of the kilowatt-hour.

Any attempt to obtain such a relatively minor saving will be warranted only if we are able to substantially reduce the duration of the cycle through limited technical effort.

4. Transportation Time

Advantage of casks containing several assemblies

It is possible to demonstrate the advantage of casks containing several assemblies, shipped by rail, with the power stations and the reprocessing plant being directly linked by rail.

The optimum solution seems to be between seven and ten assemblies per cask. As a matter of fact, above that figure the weight of the cask, with shielding of uniform thickness, exceeds 60 tons and requires the use of more expensive special rail transportation equipment. Moreover the resultant gain in the ratio of the weight of the cask to the weight of the fuel assemblies is minor.

These conclusions are valid regardless of the assumptions made about decay cooling or shipment schedule or rate.

Determination of Transportation Cost

The economic optimum (cask capacity, number of casks per train) was determined in terms of the number of reactors in operation and by considering several removal schedules or rates. This study reveals that the cask containing 10 assemblies has a slight superiority except when there is only one reactor in operation. The study also shows that it is advantageous to have three casks available as soon as the first power reactor is placed in service.

In this paper we determine the transportation cost without considering for the moment the saving obtained by a reduction in transportation time. In the table below we have summarized the two extreme solutions for casks containing 10 assemblies, loaded and unloaded at the rate of one cask per day:

the solution employing standardized (ordinary) casks, requiring a minimum of equipment, and where a single train shuttles between the operational power reactors and the reprocessing plant. The transportation cost is minimum but the transportation time increases with the number of reactors in service;

the solution employing specialized casks, and where three casks are assigned to each new reactor. The transportation cost is higher but transportation time remains uniform whatever the number of reactors in service may be.

Some 180 fuel assemblies are renewed per reactor per year.

Number of reactors	Standardized casks		Specialized casks	
	Transportation time (days)	Cost per year (10 ³ Francs)	Transportation time (days)	Cost per year (10 ³ Francs)
1	66	526	66	526
2	117	710	66	850
3	162	860	66	1180
4	216	1010	66	1500
5	256	1150	66	1830

Once several reactors are in service, reduced transportation time can be obtained only by adopting the specialized-cask solution. And yet it must be demonstrated that this choice is economically justified. For example, in the case of five power reactors the specialized-cask solution costs 680×10^3 Francs per year more than the standardized-cask solution. This extra cost reduced to the kilowatt-hour cost amounts to an increase of 0.065 percent. Yet, at the same time, the gains obtained in the time a fuel load is out of service average 3 months, or a savings of 0.6 percent in the kilowatt-hour cost. The previously mentioned supplementary study shows that this conclusion is valid whatever the number of reactors in service may be.

The most advantageous solution may, therefore, be considered to consist of: a cask containing 10 assemblies, rail transportation, employment of three shipping casks per reactor, and a transportation time of 2.2 months.

Remark

Complete specialization of the shipping casks has the advantage of guaranteeing the initiation of fuel shipment as soon as the unloaded fuel has reached the necessary decay cooling level. Power stations may consequently be shut down independently one from another without there being any repercussions on the duration of the cycle. If however, assumptions could be made on the programming of shut-downs, the same result could be obtained with partial specialization, i.e. three casks assigned to two reactors for example.

5. Regulations

We note simply that the use of sodium does not appear to be prohibited and that the temperature of the outer surface of a container can exceed 82°C provided uniformly arranged wire-guards or mesh prevent any human contact.

6. Thermal Aspects of Packaging

Studies conducted establish the following as shipment possibilities:

a. the shipment of assemblies in air makes a power of 2.5 kilowatts per assembly acceptable, provided the temperature of the outside surface of the single container can be kept at 200°C . This condition would be feasible only by having very small clearances (1 mm) between assembly and container, and with the circulation of a fluid which cools the container's outside surface. The use of helium instead of air is now under study.

b. the shipment of assemblies in sodium or lead -- with all of the cask's assemblies being housed in a same container -- makes seven to eight kilowatts per assembly acceptable. Heat is removed outside either by chimney effect with air in an open circuit, or by conduction through the shielding and radiation in the air. This latter possibility requires shielding possessing good heat conductivity. In these solutions, the inside diameter of the container is 1.5 meters for seven assemblies.

c. the shipment of assemblies in sodium or lead -- with each assembly having its own container -- makes 5 kilowatts per assembly acceptable. The heat is removed to the outside either by chimney effect with air in an open circuit, or by conduction through a matrix holding the containers, conduction through the shielding and radiation. (This latter possibility needs verification). Each container would have a diameter of approximately 250 millimeters..

d. a case similar to the preceding one, with sodium circulating inside the container by thermosiphon effect, would make more than 10 kilowatts per assembly acceptable.

Choice of Acceptable Residual Power

We shall have to compare the technical uncertainties of the different possibilities with the anticipated savings in kilowatt-hour cost.

We are able to draw up the following table, with the understanding that it shows minimum deviations, because the reference case (shipment in air) appears to be optimistic and will have to be checked.

Power at start of shipment	Reduction in fuel-cycle time	Savings in kilowatt-hour cost
2.5 kw	reference case	
5 kw	6 mos.	1.2%
7.5 kw	9 mos.	1.8%
10 kw	10.2 mos.	2 %

(see decay cooling curve in figure 2)

We note that adopting the highly cost-effective power of 5 or 7.5 kilowatts does not entail any great risks because the solutions being considered discharge the heat by conduction into the primary fluid.

In contrast, moving up to 10 kilowatts contributes but a very small additional saving, and such a move is based on the natural circulation of the sodium. Many experiments would be necessary. More complicated and more expensive containers would be required. Checks during shipment would be an absolute necessity. The expected saving is not proportionate to the effort required.

We therefore believe that thermosiphon solutions must be abandoned and that 7.5 kilowatts must be selected as the acceptable power at the start of shipment. In that case, decay-cooling time is approximately 3 months.

We also believe that shipment initiation at 5 kilowatts must be retained as a backup solution. This entails a decay-cooling time of approximately 5.6 months.

7. Shielding

Considering the temperatures to which γ shielding will be raised, we have to eliminate lead, unless we provide for its cooling. We therefore choose mainly steel (or cast iron). It would be advisable, however, to examine whether the use of uranium does not offer some advantages. And lastly, a shielding profile study, made by considering lead, shows that an appreciable weight saving (on the order of a factor of 2) is possible with regard to shielding of uniform thickness.

Neutron Shielding

The neutron shielding initially considered was a compound. Using such material with its very poor conductivity obliges resorting to solutions in which the containers are cooled by chimney effect. The cask and preparation or conditioning facilities at both the power station and reprocessing plant would be complicated thereby.

Solutions which appear worthy of consideration involve the use of cast-iron (or steel) γ shielding supplemented by a thin layer of graphite. The use of uranium shielding will also be examined.

We believe that for all solutions other than shipment in gas, lead must be abandoned because it would require cooling and substantial additional neutron shielding. Solutions utilizing steel (or cast iron) and graphite appear to have advantages.

8. Reprocessing

The chief considerations to be retained here are:

that the use of sodium as primary fluid permits tried and tested reprocessing techniques to be used, where as lead requires prior research into solvent scrubbing and separation processes;

that the storage, washing, and "trimming" of assemblies appear possible up to a residual power of 12 kilowatts per assembly;

that a solution calling for individual containers seems to facilitate unloading operations.

On the other hand, those problems presented by the presence of iodine-131 and by the possible degradation of solvents, for the residual powers being considered, have not been solved as yet.

We shall therefore conclude by emphasizing that:

it is necessary to rapidly ascertain whether these problems can be resolved for a power of 7.5 kilowatts. In the event a lower limit is determined, it would then become useless to seek ways and means of transportation at a power exceeding that limit;

verification must be made of the assumptions implicitly made in economic computations, namely:

a. reprocessing of the charge as soon as its shipment has been completed,

b. reprocessing rate higher than the arrival rate of fuels (so that fuel arriving from several power stations can be handled without inordinate waiting). A study should be made of these assumptions.

lastly, a choice must be made between lead and sodium as primary fluid.

9. Conclusion

We believe that a much earlier initiation of transportation and a decrease in transportation time make real and substantial economic savings possible, provided the rate and residual power requirements do not conflict with the rest of the fuel control chain. We further believe packaging can be designed that will make a higher residual power allowable, if there is agreement to employ a metal with a low melting point as thermal bond.

✓
ASPECTS OF INTERNATIONAL RECOMMENDATIONS
ON FAST-REACTOR FUEL SHIPMENTS

[Paper by M. Aubert; Rome, Specialist Meeting on Handling and Transportation of Spent Fuel Elements for LMFBR.
French, 26-28 April 1972, pp 1-4]

These remarks deal chiefly with the "intense sources" aspect of the 1967 edition of IAEA recommendations (Safety Series No. 6).

Proposed revisions of that 1967 edition of recommendations contain appreciable differences which are outlined below.

I - IAEA 1967 Edition of Recommendations

The following is a very concise summary of the regulations applicable to our problem:

- 1 - a leakproof containment casing is required (Annex II, Part II),
- 2 - a primary heat-transporting medium is defined (inside the containment casing),
- 3 - packages containing intense sources are subject (justiciable) to unilateral approval when 11 criteria are met (C 0.2.3.1 of the recommendations).

If these criteria are not met, the approval must be multilateral. (It should be recalled that the concept of intense source is defined by that which is above the threshold of radioactivity contained in type B packaging),

- 4 - the radiation dose rate at the surface of the packages must not exceed 200 mr/hr,
- 5 - the shipping (transportation) index is measured 1 meter from the center of the package, or for packages of large dimensions, at the end of the major axis or at 1 meter perpendicular to that major axis,

- 6 - a limit of 50 per vehicle (rail or highway) and 200 per ship, is prescribed and fixed upon the sum of the shipping indexes,
- 7 - in case of a full load, the dose rate must not exceed 200 mr/hr at all directly accessible points of the vehicle surface,
- 8 - Administrative aspects. Approval to ship and prior notification of shipment are prescribed for intense sources. This approval is unilateral when the package is unilaterally sanctioned. It is multilateral in the opposite case. In all cases, notification is given to all competent authorities of those countries through which the shipment will pass.

II - Proposed Revisions to Recommendations

A certain number of provisions are proposed for the purpose of clarifying many points of these recommendations and of simplifying them to some extent so as to preclude the possibility of misinterpretation. We consider these proposals to be a noteworthy improvement of the said recommendations.

- 1 - The new recommendations stipulate that they define actions to be taken but that responsibility for these actions is left to the initiative of governments (respective responsibilities of the consigner and the carrier).
- 2 - A leakproof containment casing is no longer defined. But what is actually defined is a containment function provided by a sealing system that cannot be opened "unintentionally". This containment must be maintained if the ambient pressure drops to 0.25 kg/cm^2 (instead of the 0.5 kg/cm^2 in IAEA 67).
- 3 - The primary heat-transporting medium has been superseded by "the radioactive content" which includes the source and the medium surrounding it.
- 4 - For that reason the recommendations define leakage rates expressed in activity rates (intensity) for both unilateral and multilateral procedures, under ordinary shipping circumstances (Annex IV. 1.2) and under accidental circumstances (tests in Annex IV. 1.4).
- 5 - The concept of intense source as defined in the 1967 edition has disappeared. We recall that the unilateral and multilateral procedures or symptoms for approval of packaging and shipment existed because of the intense sources.
Henceforth, unilateral and multilateral procedures are expressly connected with the characteristics of the type B package (leakage rate) and with the release of the source's heat under the unilateral procedure (surface temperature $\leq 50^\circ\text{C}$ or 82°C for a full load; multilateral procedures for $T_s > 82^\circ\text{C}$).

Another consequence is that the 11 criteria (C 6.2.3.1.a. of the 1967 edition) of "unilaterality" of intense sources were reduced to eight criteria relating to unilateral type B packages (in particular, compliance with the leakage rate must not depend upon the use of filters or mechanical cooling systems).

- 6 - Shipping indexes are measured 1 meter from the surface of the package.
- 7 - Dose rate at the surface: 1000 mr/hr in a full load, providing a special arrangement bars any access by unauthorized persons and the dose rate does not exceed 200 mr/hr at any point upon the surface of the vehicle, including the upper and lower surfaces (road vehicle or freight car).
- 8 - In the case of highway shipment the dose rate in the driver's cab must not exceed 2 mr/hr if the persons occupying the cab do not have dosimeters.
- 9 - The new proposals provide for shipment via freight containers. The freight container is dealt with as a package whose shipping index is the sum of the indexes of the packages contained therein. In a full load the dose rate at the surface cannot exceed 200 mr/hr (see 7 above).
- 10- A limitation of 50 is placed upon the sum of the shipping indexes per vehicle transporting packages not housed within a freight container. There is a same limit of 50 shipping indexes per freight container, except in the case of a full load where the limit is fixed by the dose rate at the surface (see 9 above).
A limit of 200 is imposed on the total of shipping indexes per ship carrying packages not placed inside freight containers.
- 11- Administrative aspects. While approvals of packages may be of the unilateral (B_u) or multilateral type (B_M), approvals to ship specifically confirm "multilaterality." Multilateral approval to ship is required for: type $B(M)$ packages with an aeration system, type $B(M)$ packages above a certain level of contained radioactivity, and packages of fissionable class III.
On the other hand, prior notification of shipment is still needed for unilaterally sanctioned $B(U)$ packages whose contents exceed a certain level of radioactivity. This notice is also required in multilateral procedures for approval to ship.

III - Fast Reactor Fuel Shipping Problems

Release of heat and the Na or NaK problem.

Principal hazards are impact upon jolting and fire. With specific reference to impact or shock it may be asked whether the prescribed drop tests at 9 meters and 1 meter are sufficiently conclusive proof of the

performance standards that may be required of such shipping casks (extremely high levels of radioactivity and hazards aggravated by the presence of Na).

Likewise, recourse to an auxiliary cooling system appears inevitable if we want to stay within regulation temperature limits, even though these limits are now more liberal. This nevertheless means that "unilaterality" is excluded and that with the present state of the art such shipments are feasible only under a multilateral system of special arrangements.

Moreover, as regards the problems associated with sodium at high temperature, we believe that the leakage-rate criteria expressed in radioactivity intensity (rates), do not cope with the possible difficulties stemming from these problems (possible leakage of Na). We further believe that in this case, technological specifications would be of such a nature as to eliminate interpretation or controversy.

We consider, however, that the introduction of special regulations for such shipments will not be feasible until we have acquired experience that confirms the soundness of these regulations. In our view, this problem is yet too new and still requires extensive developments.

For, in fact, it must not be forgotten that a regulation cannot be effective unless it can be enforced, in other words, unless it has a common-sense approach, is sufficiently simple, and is free of excessive particularity. Can it be proven that the current regulations are adequate? If so, the problem is solved. We believe that before introducing new regulations it must really be demonstrated that they are manifestly necessary; otherwise we shall run counter to the goal we seek to attain, namely, the safety of the shipments.

FRENCH EXPERIENCE IN TRANSPORTING IRRADIATED FAST REACTOR FUEL ELEMENTS

[Paper by M. Labrousse* presented before the Meeting on Handling and Transportation of LMFBR (Liquid Moderated Fast Breeder Reactor?) Spent Fuel Elements, Rome, 26-28 April 1972; French, pp 1-5, Figures 1-9, Photos 1-2, 5-9]

I. Introduction

The experimental fast reactor at Cadarache, named Rapsodie, has been operating since January 1967. During 1970 its power output was increased from 24 to 40 megawatts, the central flux reaching 3.2×10^{15} neutrons per square centimeter per second.

In it, by the end of 1971, more than 10,000 UO_2 - PuO_2 rods had been irradiated up to 10 percent content of heavy atoms. The rods were sent either to the laboratories of the CEA [(French) Atomic Energy Commission] center at Fontenay-aux-Roses for examination, or to the La Hague reprocessing plant for recovery of the enriched uranium and plutonium.

II. Containers

Among these rods are some with broken sheaths, either accidentally sustained in the pile or in the course of handling, or else deliberate. In order to transport them, several methods of packaging and several types of containers were used.

Nominally Whole Fissionable Rods

The stacks comprising 37 rods are dismantled in a cell and the rods sent to the pilot plant at La Hague.

Let us recall that these rods are 6.7 mm in diameter (7.9 with the spacer wire), 485 mm long, and contain (before irradiation) 20 grams of

* CEA Production Directorate, Subdirectorate for Plutonium.

fissionable Pu and 34 grams of U-235. They are placed in massive racks made of AG1 [designation for type of steel] (Fig. 1, Photo 1) providing effective cooling, each of which can hold 116 of them, or about three stacks ($3 \times 37 = 111$). Each rack is enclosed in a case made of AG3 [designation for type of steel] (Fig. 1, Photo 1) 2.5 mm thick, with closure cold-welded by pressure in the cell. Tests have proved that this closure remains airtight under internal pressure of 3 bars, but is not subject to regular verification inspection. It does not enter into analysis of the confinement provided by the container, even though it obviously provides additional safety; its primary purpose is to avoid contamination of the loading and unloading cells.

Three of these cases are placed in the three receptacles of a type IU 07, called the Rapsodie 3-receptacle container (Fig. 2, Photo 2). Each receptacle, lined with 10-mm thick stainless steel, is within a block of lead 900 mm in diameter (minimum lead thickness, 211 mm) and is closed with a stainless steel-sheathed plug of 225-mm thick lead. The three plugs are held in place by a stainless steel lid, 20 mm average thickness, fastened with 12 bolts, 16-mm diameter.

The airtight seal is duplex; it is provided, first, by the annular joint on each of the plugs, and second, by one on the lid.

This lid is covered by a hood consisting of two dished facings of A 42 steel, 6 mm thick, with 59-mm thick heat insulation between them.

The body of the container consists of two concentric A 42 steel shells, the outer one 6 mm thick and the inner one 26 mm thick, with 33-mm thick heat insulation between them.

The total empty weight (without frame and lugs for securing) is 8.5 tons.

Overall dimensions are: Height -- 1,385 mm
Diameter -- 1,330 mm.

Since the reactor power was increased, new stacks have been placed into service; they contain 61 rods, which are both longer (532 mm instead of 485) and thinner (5.1 mm without the spacer wire and 5.92 with it).

The same containers and cases are used; only the rack is different. Its usable height is 517 mm (instead of 470) and its overall length is 651 mm (compared with 545); it has 152 holes instead of 116.

The amount of fissionable material per rod is 10 grams of Pu and 25 grams of U-235.

The rated power of both these types of rods should not permit 1,600 watts to be exceeded, or 4.8 watts per rod for the first type and 3.5 watts per rod for the second, which correspond to deactivation periods on the order of 100 days.

Containers are always handled in vertical position. They are placed beneath the loading or unloading cells after the hood and lid have been removed. The cell closure and receptacle plugs are lifted into the interior of the cell. Unloading the three receptacles requires 120° rotation of the container, with each receptacle replugged before being rotated out of position. No special provision for airtightness between container and cell is made; the loading and unloading cells are in fact separated from the others (dismantling and dissolving) and only cases which are not contaminated are handled there.

This system, which is rather special, obviously complicates the design and increases the cost of the cells; on the other hand, it greatly simplifies the maneuvers and avoids any contamination. For the Phenix reactor, where the stacks will be dismantled -- at least in the beginning -- there will be no uncontaminated loading cell available, and airtightness between cell and container, to be accomplished before the lid is raised, must be provided.

Broken or Cut Rods

These are first sent to the RM II laboratory at Fontenay-aux Roses for examination and then to the La Hague plant. Initially it was decided to transport all these rods in an airtight container. The system described in the foregoing was retained, but the rack contained in the case is called a "light" rack, consisting of four aluminum discs welded to a central shaft and drilled with a certain number of holes (30 or 36) (Fig. 3).

The cases (Fig. 4), whose usable diameters may vary from 10 to 14 mm and usable lengths from 488 to 817 mm, are made of stainless steel, closed with a screw-plug with a socket for a hexagonal [Allen] wrench.

The container is constructed upon the same principle as the preceding, but, of course, with a single receptacle instead of three. The lateral thickness of lead is 220 mm, which is constant, and the "calorific capacity" (considering the temperatures) can be as high as 1,800 watts. The main difference is a single external shell 10 mm thick and the use of plaster as heat insulation.

The two types IL 24 and IL 25 differ only in height (and weight):

	IL 24	IL 25
Usable height	710 mm	910 mm
Overall height	1,496 mm	1,697 mm
Weight	3,800 kg	4,500 kg

and by the fact that the shell of the second is made of stainless steel which enables it to be submerged in a pool without fear of corrosion.

It soon became apparent that the use of cases in certain instances was an unnecessary luxury, both from the standpoint of safety and practical operation. For one thing, tests simulating accidents proved that no water could penetrate into the interior of the container; for another, experience has proved that rods whose sheaths have been only slightly damaged (normal sheath breaks) presented no risk at all of fuel dispersion. Thus we use the cases only for rods which have been cut or seriously damaged, when there is risk of fuel escaping from the sheath.

In certain special instances, rods can be transported in types IL 19, 20, 21, and 22 containers (Figs. 7,8, and 9; Photos 7,8, and 9), all of which have the following features:

- One receptacle with specimen carrier handled by means of a bar;
- Horizontal unloading;
- Screw-operated guillotine closure; and
- Thickness of lead -- 240 mm.

Containers of these types are currently being used for examination specimens. The last three differ only in length and weight.

Feeder Elements

These present little problem because they contain much less radioactive material than fuel elements proper. We have been using an existing container, built for graphite-gas fuel elements.

III. Transportation Methods

The foregoing containers, since their weight does not exceed 20 tons, can be shipped either by highway or by rail. The latter is slower but, despite highway transport terminals -- since neither Cadarache nor La Hague are on routes -- proves to be more economical.

IV. Conclusion

Several thousand highly irradiated rods have been transported practically from one end of France to the other with no problem whatever, and, of course, without accident, or even minor incident. We must nevertheless be aware that the conditions of this transportation are rather different from those which have to be observed with irradiated fuel elements of commercial reactors. To resolve problems of criticality and temperature at the same time, we are compelled to transport stacks under sodium, which is a dangerous material. To remove the heat will no doubt necessitate, as with the containers for light water fuel elements, systems for artificial cooling. The presence of sodium and a highly toxic fissionable material, and the possibility, in the event of accident, of loss of cooling, will require confinement precautions still more severe than in the cases of graphite-gas and light water fuel elements. But these precautions will undoubtedly be easier to observe since it does not appear that the desirability of having containers of very large capacity is as great as in the latter two cases.

Keys to Figures:

[Notes for all figures --

Dimensions are in millimeters.

Symbol ϕ means diameter.

CEA DP Section des Transports Speciaux = CEA Production
Directorate, Special Transport Section]

Figure 1:

1. Designation : AA 55
Nomenclature : Handling Device for Rapsodie
Fissionable Rods (Whole)
2. Shape : cylindrical
3. Next assembly : IU 07
4. Weight : 20.485 kg
5. Volume
6. Holders
7. Number
8. Makers
9. Date
10. Note : 116 holes, 9.5 mm dia.
11. Cold weld
12. Airtight bottom piece

Figure 2:

1. Section A-A
2. Heat insulation
3. Annular joint; 7 mm dia. cross section
4. Ring joint
5. 2 lifting lugs
6. Lead flakes + plaster
7. 10-mm thick stainless steel
8. Usable diameter 169 mm
9. 211-mm thick lead
10. Usable height 710 mm
11. 1,090 mm between pins
12. 2 alignment brackets
13. 4 tie-down lugs; 60 mm dia., A 48 steel pin
14. Lid fastened with 12 bolts, 16 mm dia.
15. In accordance with Lemer drawing 57471B, file 6727

Figure 3:

1. Nomenclature : Light racks for Rapsodie reactor fissionable rods
2. Designation
3. Usable diameter
4. Weight
5. Drawing number
6. Number of holes
7. 3 braces, 8 mm dia., @ 120°

Figure 4:

1. Nomenclature : Cases for cut Rapsodie reactor rods
2. Reference
3. Case dimensions
4. Weight
5. Drawing No.
6. Hexagonal socket head
7. Across flats
8. 1-mm thick stainless steel
9. Usable height
10. Overall height
11. Usable diameter
12. Outside diameter

Figure 5:

1. Section A-A
2. Shackle
3. Viton joint
4. 4 hex-head bolts, 18 mm dia.
5. Plaster
6. 6 hex-head bolts, 18 mm dia.
7. Usable height 710 mm
8. Usable diameter 169 mm
9. Stainless steel, 7 mm thick
10. View at F
11. Tie-down lug
12. Top view -- half of lid removed
13. According to SLPI drawing 6531-0A

Figure 6:

1. Section A-A
2. Shackle
3. Viton joint
4. 4 hex-head bolts, 18 mm dia.
5. 6 hex-head bolts, 18 mm dia.
6. Plaster
7. Usable height 910 mm

[Key to Fig. 6 continued on p. 7]

[Key to Figure 6 continued from p. 6]:

8. Usable diameter 169 mm
9. Stainless steel, 7 mm thick
10. Stainless steel, 10 mm thick
11. Plaster, 10 mm thick
12. Stainless steel, 20 mm thick
13. View at F
14. Tie-down lug
15. Top view -- half of lid removed
16. According to SLPI drawing 8262-00

Figure 7:

1. Section A-A
2. Metal-plastic joint
3. Specimen carrier
4. Usable diameter 82 mm
5. Usable length 600 mm
6. Lid fixed by 6 hex socket head bolts, 16 mm dia.
7. Tapped, 20 mm metric thread
8. 6 hex socket head bolts, 14 mm dia.
9. 6 hex socket head bolts, 10 mm dia.
10. Plaster
11. Tapped, 40 mm metric thread
12. 2 hex head bolts, 24 mm dia.
13. Section B-B
14. 4 hex head bolts, 30 mm dia.
15. A 42 steel, 19 mm thick
16. Stainless steel, 7.5 mm thick
17. According to OTM drawing 124-M235 B

Figure 8:

1. Lug -- 31 mm dia.
2. Section A-A
3. Outer cover -- 4 hex head bolts, 16 mm dia.
4. Specimen carrier -- usable diameter 85 mm
usable length 600 mm
5. Plaster
6. 6 hex head bolts, 16 mm dia.
7. 2 alignment pins
8. Silicone joint
9. 10 hex head bolts, 16 mm dia.
10. Alignment with cell on this diameter -- 630 mm
11. Section B-B
12. 4 handling lugs, 31 mm dia.
13. Stainless steel, 7.5 mm thick
14. Adx steel, 12 mm thick
15. Plaster, 10 mm thick
16. According to SLPI drawings N 8230-00 and N 8230-08

Figure 9:

1. Lug -- 31 mm dia.
2. Section A-A
3. Outer cover -- 4 hex head bolts, 16 mm dia.
4. Specimen carrier -- usable diameter 85 mm
usable length 1,300 mm
5. Plaster
6. 6 hex head bolts, 16 mm dia.
7. 2 alignment pins
8. Silicone joint
9. 10 hex head bolts, 16 mm dia.
10. Section B-B
11. 4 handling lugs, 31 mm dia.
12. Stainless steel, 7.5 mm thick
13. Adx steel, 12 mm thick
14. Plaster, 10 mm thick
15. According to SLPI drawing N 7157-0

(1)

AA.55**AA.55****DESIGNATION****dit Amenagement pour aiguilles fissiles Rapsodie (Saines)****FORME: Cylindrique (2)****MASSE 20,485 Kg (4)****VOLUME (5)****COMPLETE IU.07 (3)**

(6) Detenteurs	(7) Nombre

(8) Fabricants	Nombre	(9) date
AA-138 : Desmouceaux		
AA-139 : GFR/CEN/Co		

OBSERVATIONS116 trous $\varnothing$ 9,5 (10) **Figure 1**

(Key on page 48)

AA-55

CEA DP Section des Transports Speciaux

IU.07

(1) Coupe A-A

(12) pattes de centrage

Figure 2

D'après plan Lemer 574718 dossier 6727 (15)

(Key on page 48)

IU-07

Photo 2

IL-25

AA-162 AA-163

(1) dits rateliers allèges pour aiguilles fissiles du réacteur RAPSODIE

Figure 3

(Key on page 49)

IL-25

AA-141 AA-148 AA-153 AA-156 AA-159

(1) dit etuis pour aiguilles tronconnées du reacteur RAPSODIE

(2) REFERENCE	(3) DIMENSIONS ETUI					(4) POIDS	(5) N° DE PLAN
	Du	Hu	De	He	x		
AA-141	10	488	12	500	8	140 g	592RAM1008 2.2
AA-148	12	646	14	655	8	215 g	ACB J.183.00.15
AA-156	10	535	12	544	6	155 g	ACB J.183.00.27
AA-159	12	813	14	822	8	270 g	ACB J.183.00.18
AA-153	14	817	16	828	8	310 g	ACB J.183.00.08

Figure 4

(Key on page 49)

IL-24

(1) Coupe A-A

(10) Vue suivant F

Vue de dessus (12)
½ couvercle enlevé

Figure 5

(Key on page 49)

D'après plan SLPI 6531-0 A (13)

IL-24

Photo 5

IL.25

(1) Coupe A-A

(13) Vue suivant F

Figure 6

[Key on pp 6,7]

D'après plan SLPI 8262-00 (16)

IL-25

IL-19

(1) Coupe A-A

Figure 7

D'après plan OTM 124-M235 B (17)

(Key on page 50)

IL-19

CEA-DR Section des Transports Speciaux

CEA-DR Section des Transports Speciaux

IL-20

(10) Centrage pour cellule $\varnothing 630$

Figure 8

D'après plan SLPI N 8230-00 et N.8230-08 (16)

(Key on page 50)

IL-20

CEA DP Section des Transports Solides

CEA DP Section des Transports Solides

IL-22

Figure 9

D'après plan SLPI N 7157-0 (15)

(Key on page 51)

IL-22

✓ PROBLEMS IN RECEIVING FAST REACTOR FUEL ELEMENTS AT REPROCESSING PLANTS

[Paper by B. Lenail* presented before the Meeting on Handling and Transportation of LMFBR (Liquid Moderated Fast Breeder Reactor?) Spent Fuel Elements, Rome, 26-28 April 1972; French, pp 1-4]

In spite of the fact that fuels irradiated in fast reactors have been transported for several years, it is still difficult to generalize about the problems which are, or will be, encountered in receiving irradiated fuels in the next 5 or 10 years.

This is related, on the one hand, to the fact that the solutions which have already been tested will not be used any longer, and on the other, to the fact that the solutions already planned for Phenix will resolve the problem only during the first few years of reactor operation (transport of assemblies dismantled into rods freed of sodium).

The following three cases will be reviewed in sequence:

Receiving of Rapsodie and Fortissimo fuel elements at AT 1 (La Hague);

Receiving of Phenix fuel elements (without sodium) at UP₂-HAO [High Oxide Activity] (La Hague); and

Receiving of fuel elements transported under sodium.

I. Rapsodie-Fortissimo Rods Delivered to AT 1

The fuel elements are delivered to AT 1 either in the form of rods, considered whole, or in the form of rods which have been broken or cut up into pieces, placed in separate containers.

In either case, transportation as well as unloading is done dry.

* CEA [(French) Atomic Energy Commission] Production Directorate, Subdirectorate for Plutonium.

Unloading is accomplished by connecting the opening of the transport container to a cell.

Upon arriving, the boxes containing the rods enter, by remote control, into a γ -protected receiving cell, and then into the shearing cell which is α , β , and γ -protected. In a series of remote-controlled or telemanipulated operations:

A box is opened (by means of a rotary wheel cutter);

The rack is removed, and stored in a protected case;

One by one, rods are taken up by the telemanipulator and placed in the shear; and

The rods are cut in a vertical rotary blade shear into 10-mm pieces which fall into the dissolver basket.

Problems encountered in the use of such a system are very much limited because essentially very little contamination is being dispersed, and the confinement of radioactive substances is broken only after they have entered the cells.

In fact, the only sources of contamination are gaseous fission products, and the very low danger arising from them is well known.

Another problem, which it might have been feared would also raise some difficulties, is the possibility of sodium being present on the rods despite the cleaning performed before shipment. The amount of sodium present on the surface of each rod (about 41 grams of plutonium each) is estimated at less than 1 gram, and at this level no difficulty at all can be observed, either during opening of the containers, or shearing the rods, or introducing the cut pellets into the dissolver.

It is appropriate to observe in this connection that radial cover assemblies, after being similarly cleaned of sodium, have been immersed in water with no difficulties at all.

II. Phenix Assemblies Freed of Sodium

Starting in 1974, or in 1975 at the very latest, the irradiated fuel elements unloaded from Phenix will be sent to La Hague for reprocessing.

In fact, the plant now used for processing graphite-gas fuel elements will at that time be outfitted with the necessary equipment for processing LWR [light water reactor?] and FBR [fast breeder reactor?] fuel elements.

At least during the first few years of operation of the reactor, it has been decided to effect both the removal of the sodium and dismantling of the stocks before shipment. This choice was made far less because of essential conditions peculiar to industrial processing of these fuel elements than to satisfy the requirements for numerous examinations of irradiated rods in this prototype reactor.

For shipment, the rods are grouped in bundles, four of which are placed in a specially designed container.

At the La Hague plant the containers are submerged and unloaded in a pool (demineralized water) in the conventional installation for receiving LWR fuel elements.

No difficulty peculiar to this type of element need be feared for the present in this area, since in 1975 there will be the benefit of the experience acquired in handling power reactor elements (gas or ordinary water).

III. Assemblies Under Sodium

It is practically certain that for technical reasons (emptying high-power reactors) or economic (avoiding the necessity of different centers for stocking both before and after shipment) we will be led to shipping fuel elements for commercial power reactors (1,000 to 1,200 megawatts electrical output) under sodium, the entire assembly along with the sodium being placed in an airtight flask, one or several of which will be placed inside the shipping container.

Even though the method of loading the shipping containers is now known (dry, from a cell), the question of unloading the shipping containers has not yet been studied because:

Either, after 1985, it will involve specific plants for which solutions much more closely tied to what will be taking place in the reactors will be employed,

Or else, in an intermediate phase (1980-1985), it will involve multipurpose (LWR/FBR) plants and the solutions now employed (stocking under water in a pool) will be poorly suited to unloading and stocking fuel elements packaged in sodium-filled flasks.

It is conceivable that by 1980 the choice of shipping under sodium will be a forced one in France, and perhaps even sooner, if the practice of disassembling in cells near the reactor is abandoned at Phenix.

It will then undoubtedly be necessary to supplement the equipment of the La Hague plant to enable it to meet this new requirement (unloading pit with equipment for sodium removal).

IV. Other Problems To Be Resolved

As can be seen, the problems in receiving and unloading shipping containers will not be especially difficult ones with fast reactor fuel elements. However, in addition to the above mentioned difficulties when the containers loaded with sodium are received, the other operations constituting the mechanical preprocessing can be delicate. Such operations are closely dependent upon the processing procedure selected; here it will have to suffice to enumerate the sequence of operations encountered in aqueous processing:

Removal of sodium with steam;

Sawing off the heads and bases of the stacks (possibly removal of structural members);

Dismantling the stacks into rods, bundles, or modules small enough to be safely handled in the air; and

Shearing the rods or bundles and separating the sections of sheath filled with oxide to be fed into the dissolver from the fragments of auxiliary parts (gratings, wires, springs, and so on).

Most of these operations are fairly similar to those used with LWR elements; nevertheless, serious complications may well be feared because of:

Much greater residual activity (with equal cooling, greater than that of LWR fuel elements by a factor of 4 to 6);

The presence of sodium in the stacks or around the rods, and inside broken or cracked rods, or those with holes;

The high content of gaseous fission products (greater than that of LWR fuel elements by a factor of 2-3);

The extreme fragility of highly irradiated sheaths; and

Possible criticality problems because of the high -- 15 to 20 percent -- content of fissionable materials.

INTERNATIONAL ATOMIC ENERGY AGENCY

SPECIALIST MEETING ON HANDLING AND TRANSPORTATION OF
LMFBR SPENT FUEL ELEMENTS

Rome, 26-28 Avril 1972

✓
EXPERIMENTAL TRANSPORT OF DUMMY FUEL
SUBASSEMBLIES FOR "JOYO"

by

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- JAPAN

The experimental fast reactor, "JOYO", which is the first sodium cooled fast reactor in Japan, is expected to reach criticality in 1974. Accordingly, we have so far accumulated none of practical experiences in handling and transporting the spent fuel from the fast reactor. However, with regard to the handling and transport of the new fuel for "JOYO", some R & D works have been made using dummy fuel subassemblies. We will investigate in the near future the problems associated with transporting spent fuels which are anticipated to arise around 1975.

Mentioned briefly hereunder are the results of the transport experiments of dummy fuel subassemblies for "JOYO". Both core and blanket fuels of "JOYO" are of herical wire type. In the experiments for core subassemblies, however, two kinds of dummy core fuels, respectively herical and grid types, were prepared and tested. They were transported by a truck for a distance of approximately 30 km from the Pu fuel fabrication plant at Tokai to the "JOYO" site at Oarai. Vibrating characteristics of various parts of the subassemblies were observed and degree of deformation was measured after transportation. Important results obtained by the experiment are summerized as follows:

- (1) For herical wire type pin, major resonance was observed at excitation frequency of 50 - 60 Hz. In the case of grid type pin, major vibration takes place at 50 Hz and also 300 Hz.
- (2) Transport was made at the speed of approximately 40 km/h. The mean value of acceleration observed on herical wire type pin was approximately two times the mean acceleration observed on the truck body. For grid type pin, the factor exceeded three.
- (3) The stress produced at various parts of the fuel subassemblies during transportation was estimated to be sufficiently smaller than the allowable stress. No significant deformation was observed after transportation. Damage due to wearing was so small that it could be regarded as negligible.

A similar experiment was made with a dummy blanket fuel subassembly containing herical wire type pins. Nearly the same results were obtained as in the case of the core fuel subassemblies, except for the resonance frequencies which were observed to be 40 Hz and also 70 Hz.

In addition, a conceptual design of a transport cask for new fuel subassemblies for "JOYO" has been made.

In general, transport of nuclear fuel in Japan is controlled by the Law for Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors. As to the design criteria for transport casks, current practice is based on the guide produced from the IAEA Regulations and 10 CFR part 71 of the USAEC.

UKAEA KINGDOM ATOMIC ENERGY AUTHORITY

✓ TRANSPORTATION OF LMFER FUEL ELEMENTS

K J HENRY

FOR DISCUSSION AT THE IAEA SPECIALIST MEETING - ROME, 26-28 APRIL 1972

UKAEA
REACTOR GROUP
RISLEY
WARRINGTON
LANCS

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TRANSPORTATION OF LMFBR SPENT FUEL ELEMENTS

The two British Fast Reactors - The 60 MW(TH) DFR, which has been operational for the past 12 years; and the 600 MW(TH) PFR, which is nearing completion of construction - are both situated at the Dounreay site of the United Kingdom Atomic Energy Authority.

Dounreay is an integrated Fast Reactor establishment in that it has fuel manufacturing and reprocessing plants. The DFR fuel is manufactured and reprocessed there, PFR fuel will be manufactured by British Nuclear Fuels Ltd, but for the next decade until the Civil Fast Reactor Fuel Reprocessing Plants come on line the PFR fuel elements will be reprocessed at Dounreay. Thus no major problem yet exists of transporting large quantities of irradiated LMFBR fuel elements on public roads and railways within the British Isles.

The need to have available a LMFBR spent fuel transport system will not arise in the UK until the latter part of this decade when the first 1300MW(E) Civil Fast Reactor, now being designed comes on line. Nevertheless, the overall safety appraisal of this reactor is being studied now, and therefore the UKAEA in conjunction with the CEEGB and BNFL are carrying out a design study of a transport flask and are planning a series of tests in accordance with the IAEA regulations.

A flask specification has been prepared with the following main requirements:-

1. It must conform with the IAEA Regulations for the safe transport of radioactive materials, flask surface temperatures, radiation levels, fire resistance, and resistance to impact damage.
2. Heat transfer will be by natural conduction and convection, ie no forced cooling arrangements to be provided, and pin temperatures must not exceed 650°C under normal conditions and 700°C under fire accident conditions.
3. The flask must be capable of being carried by road and rail within the British Isles. This implies a weight limit of about 100 tonnes, a maximum length of 7 m, a maximum diameter of 2.7 m over projections.
4. The flask should be compatible with the handling arrangements of the PFR fuel handling caves, and the proposed CFR fuel handling scheme.

Within the above constraints, a design has evolved which is capable of carrying 5 sub-assemblies with a total heat rating of about 40 KW. Gamma shielding is provided by 0.48 m thickness of steel; the material of the external neutron shield has not yet been fixed, but a water annulus or a boron impregnated resin of equivalent attenuating capacity are both under consideration.

The sub-assemblies are contained in sodium filled canisters 220 mm dia x 4 m long x 12 mm wall thickness, with seal welded end caps. These fit into separate cylindrical compartments in the flask which are in turn covered by bolted flanges with double testable seals. A further bolted flange with double sealing rings is also provided.

SHIELDING REQUIREMENTS

The gamma emission from irradiated fuel depends mainly on the rating and decay time and although the fission product yields differ for fast and thermal fission, and for fission of the various isotopes 235, 238, 239, the total gamma activity does not vary greatly. Satisfactory data exists, therefore, for calculating gamma shield thicknesses for the fuel loading specified. The neutron source strength depends on the isotopic composition of the fuel and in particular on the Plutonium 241 content.

During reprocessing curium and americium are removed from the fuel, but thereafter the americium content grows steadily as a result of Pu 241 decay. During subsequent irradiation curium 242 and curium 244 are produced; these are the major neutron emitters.

In choosing an isotopic composition therefore as a basis for design studies a figure for the storage time between reprocessing and irradiation has to be assumed. The first British Civil Fast reactors will be fuelled with Plutonium from Magnox and Advanced Gas Cooled Reactors. It is considered unrealistic to design for fuel that has been stored, for say, longer than 4 years. This figure has therefore been assumed together with a burn-up of 10%.

The activities calculated for CFR sub-assemblies irradiated to 10% burn-up give rise to shielding requirements of about 500 mm steel for gamma shielding and a 76 mm thick water annulus or equivalent for neutrons.

HEAT TRANSFER AND TEMPERATURE CONDITIONS

The IAEA regulations impose two main temperature limitations, first that the temperature of the surfaces of the flask at any accessible point shall not exceed 82°C, and secondly that under the conditions of an external fire in which the temperature of the environment is raised to 800°C and maintained for 30 minutes, followed by a return to an environmental temperature of 26°C, the contents of the flask should not exceed safe temperature limits.

In the case under discussion a maximum fuel pin temperature of 700°C has been imposed under steady state and fire conditions. Temperature distributions have been calculated for this flask arrangement for both conditions with varying heat loads - Figs 1, 2 & 3.

The first steady state temperature limit reached is that of the flask surface at 82°C, which results from a heat loading of about 40 KW. If access to the centre portion of the flask is prevented, for example, by fitting a protective grid, then the heat loading can be increased to about 45 KW before reaching the fuel pin temperature limitation.

The effect on heat transfer of the insulating barrier imposed on the outer surface of the flask by the neutron shield can be reduced to a certain extent by allowing every third fin on the outer surface of the neutron shield region to pass through to the steel surface of the flask.

Under the conditions of the IAEA fire test the maximum fuel temperature reached with the 40MW heat loading is about 630°C. Under these conditions of course the neutron shield, if water, would evaporate and escape via a bursting disc or safety valve. In the case of a solid neutron shield this would burn and be automatically removed. Following the removal of the neutron shield the neutron dose rate at the surface would be increased tenfold. This situation is of course catered for in the IAEA regulations which permit an increase of 10^2 in the radiation levels from the flask surface under accident conditions.

The free volume of gas remaining in the canister after seal welding (this operation will be carried out, it is assumed, at an ambient flask temperature of about 150°) will be adjusted to ensure that if the fission product gas content of all the pins in a sub-assembly at a temperature of 700°C, the stresses imposed on the canister will be in a safe range.

There is a radial gap of about 6 mm between the canister and its pocket. The quantity of sodium in, and hence the dimensions of the canister and its pocket will be arranged, to ensure adequate heat transfer in the event of a leak occurring in the canister which allows sodium to escape into the pocket.

TESTING PROGRAMME

Following upon the design study, a drop testing programme has been arranged in accordance with the procedures laid down in the IAEA regulations. These require a demonstration that the flask is capable of withstanding a free fall from a height of 30 ft onto a "rigid" target. The fall being arranged in such a manner that the "weakest" area of the flask is subjected to the impact.

In recent years a variety of thermal reactor spent fuel transport flasks have been designed to these requirements in the UK. The design of these has been based on data extrapolated from numerous tests on $\frac{1}{16}$ and $\frac{1}{4}$ scale models, since acceptable methods exist for correlating the test data to the behaviour of full scale flasks.

The CFR sub-assemblies unlike those from thermal reactors are doubly contained, and it is not considered feasible to scale both canisters and flasks and to then extrapolate the results to full size. It has therefore been decided to subject full size canisters to the IAEA drop test to ensure that they are independently capable of withstanding the specified impacts.

The programme will therefore consist of a series of tests on $\frac{1}{4}$ scale models to enable deceleration rates, and hence shock absorber details, to be determined, followed by tests on the canisters with the same decelerating rates applied. In order to simulate the support against buckling afforded to the canister by the flask body, the test canisters will be enclosed in thick walled cylinders.

Since the canisters will be required to withstand the stresses from impact at temperatures in the region of 700°C, the deceleration rates will be arranged so that the stresses in the canister wall and buckling forces are within a range which takes account of the reduced strength of the material at this high temperature.

These, then are the proposed tests which conform to the IAEA regulations. There is a body of opinion in the UK however, which feels that the IAEA tests may not represent the maximum impact hazard which may arise from train accidents at the very high speeds proposed for the next decade. Discussions on this aspect are at present taking place between the various organisations concerned in the United Kingdom.

CONCLUSIONS

The design study for the CFR Irradiated Fuel Transport Flask has proceeded to the stage where it is considered feasible to provide a design capable of carrying 5 CFR type irradiated fuel sub-assemblies with a total heat rating of 40 KW, and which will meet the requirements of the current type B container transport regulations.

During the above design study, however, a series of questions have arisen which may merit further discussion at the meeting.

1. In view of the high decay heat powers of LMFBR spent fuel, should the provision of forced cooling to the flasks be considered, with the consequent safety requirements of guaranteed power supplies, permanent operating staff, etc.
2. In order to reduce the possible hazard which may arise from transporting this fuel over public transport systems should some form of head-end treatment be carried out at the power station to remove, for example, volatile fission products, reduce the bulk of the fuel element material, reduce heat output, and produce a more inert and probably safer material for transport.
3. Because of the intention of most western countries to develop railway systems capable of operating at speeds well in excess of 160 kph, are the present IAEA test regulations stringent enough to cope with impact forces which could arise from train accidents at this speed.

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5 April 1972

FIG.1. FIRE TRANSIENT

FIG.2. CFR FLASK STEADY STATE TEMPERATURES

FIG. 3. FLASK SURFACE TEMPERATURE DISTRIBUTION