Development of a MOX Fresh Fuel Shipping Cask for Light Water Reactors*

T. Meguro[1] and T. Ohtake[2]

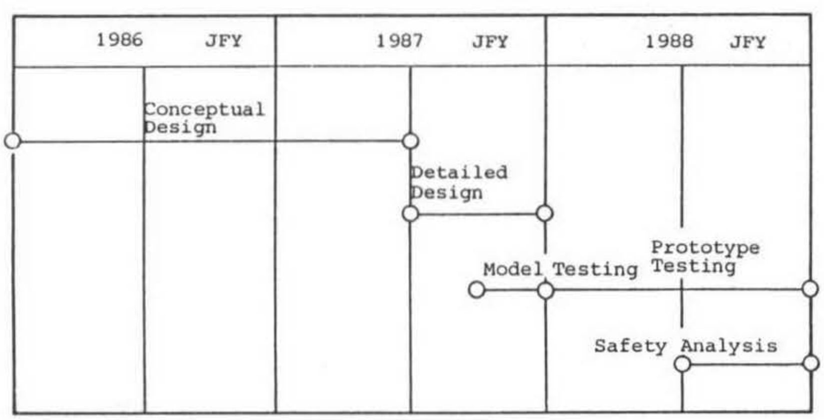
[1]*The Japan Atomic Power Company, Tokyo*
[2]*Power Reactor and Nuclear Fuel Development Corporation, Tokyo, Japan*

1. Introduction

It is very important to use plutonium for light water reactor (LWR) fuels from the standpoint of one of the effective use of recovered plutonium. Progress is being made in Japan on a plan to use uranium-plutonium mixed oxide (MOX) in LWRs as "Pu-thermal" program. BWR/PWR common usable MOX fresh fuel shipping package has been developed in the last three years to transport MOX fresh fuel assemblies from MOX fuel fabrication facility to the sites of LWR where they will be used.
This paper describes the outline of the development.

2. Background of Development

It is required to develop a packaging which can transport a plenty of MOX fresh fuel assemblies for LWRs, safely and economically. This program to develop this type of package has been successfully performed in accordance with the following schedule:



* This work is being done under contract between the ten Japanese electric power companies and the five participating companies:

The Japan Atomic Power Co.; The Hokkaido Electric Power Co., Inc.; Tohoku Electric Power Co., Inc.; The Tokyo Electric Power Co., Inc.; Chubu Electric Power Co., Inc.; The Hokuriku Electric Power Co., Inc.; The Kansai Electric Power Co., Inc.; The Chugoku Electric Power Co., Inc.; Shikoku Electric Power Co., Inc.; Kyushu Electric Power Co., Inc.; Power Reactor and Nuclear Fuel Development Corp.; Toshiba Corp.; Hitachi, Ltd.; Mitsubishi Heavy Industries, Ltd.; and Nuclear Fuel Industries, Ltd.

3. General Characteristics and Specification of the Package

This package, designed as package for type BM with fissile class I, can contain two (2) PWR type fresh fuel assemblies or six (6) BWR type fresh fuel bundles by exchanging the internal supports which can assure the integrity of the fuels in transport. This package is in cylindrical shape, and the specifications (dimensions, weight, the materials of main components) are shown in Fig.3-1 and Table 3-1.
And the main characteristics are described as follows:

(1) Installation of inner fins for sufficient decay heat dissipation
(2) Capability of handling it both horizontal and upright ways taking into account of fuel fabrication facility and Japanese nuclear power stations.
(3) Capability of using the existing LWR fuel transport systems in Japan.

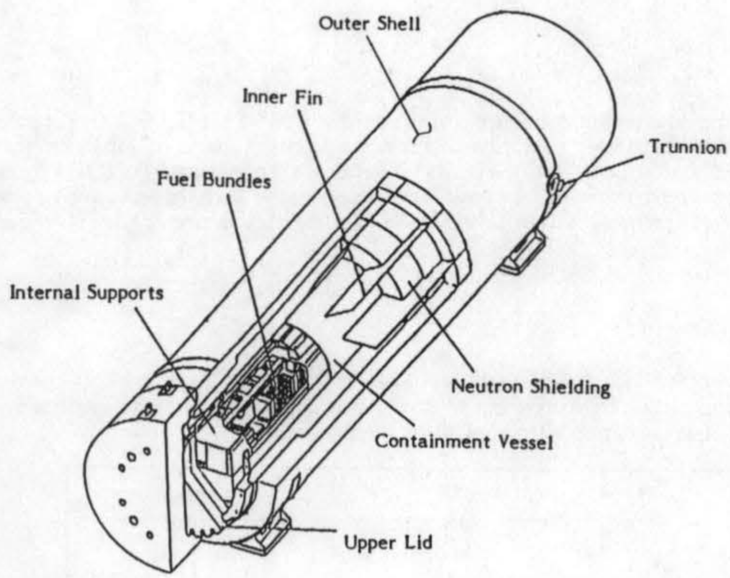


Fig. 3-1 General view of the package (BWR type)

Table 3-1 Specification of the package

	PWR	BWR
No. of Accomodated Fuels	2	6
Weight of Package (MT)	8.2	9.1
Size of Package (m) · Outer Diameter · Whole Length	Approx. 1.3 Approx. 5.5	
Main Materials · Structure · Shock Absorber · Neutron Shielding	Stainless Steel Balsa Wood Silicon Rubber (B_4C Contained)	

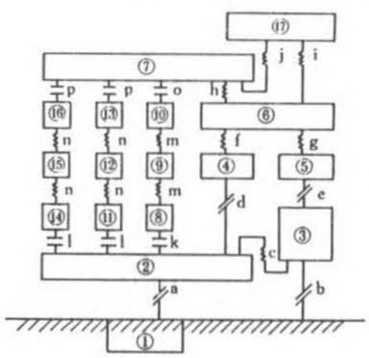
4. Outline of the Safety Analysis

The package has been confirmed as type B package by evaluating structure, thermal, containment, shielding and subcriticality under the normal and accident test conditions which are defined by the Regulation. And the adequacy of evaluation method has been benchmarked by comparing the evaluations and the experiments described in Chapter 5.

4.1 Structural Analysis

It is confirmed that the integrity of the container is maintained structurally under the regulatory conditions. The validity of a dynamic analysis method using a spring mass model (SHOCK-code), an example is shown in Fig. 4.1-1, can be confirmed as a proper method to evaluate acceleration and deformation caused by 9m drop (accident) and 0.9m drop (normal), because it is clarified that the above analysis gives conservative values as the results of the comparison between the evaluation and the experiment shown in Table 4.1-1.

Fig. 4.1-1 Spring-mass system model (Vertical Drop)



Nα	Portions
①	Unyielding Target
②~⑦ ⑰	Main Body
⑧~⑯	Container's Internals

～ : Linear Coupling

—||— : 〃 (Compression only)

—//— : Non-linear Coupling

Table 4.1-1 Comparison between the evaluation and the experiment

		1/3 Scale Model		Prototype Model	
		Evaluation	Experiment	Evaluation	Experiment
Vertical	Acceleration (G)	750	691	370	300
Drop	Deformation (mm)	21	15	39	23.6
Horizontal	Acceleration (G)	480	349	183	120
Drop	Deformation (mm)	30	22	96	94.5
Corner *	Acceleration (G)	250	198		
Drop	Deformation (mm)	75	75		

* Only the experiment by 1/3 scale model was executed.

4.2 Thermal Analysis

The thermal analysis using the "TRUMP" code was performed to verify the thermal characteristics of the package under the normal and accident test conditions. Analytical methods have been confirmed by comparison between the results of the thermal analysis and the thermal tests using slice models and a prototype model. As the result of the thermal analysis, the temperatures of various parts of the package are acceptable for the materials and meet the requirements of the Regulations, as shown in Table 4.2-1.

Table 4.2-1 Maximum calculated temperatures of the package

Condition	Part	Maximum temperature (°C)	
		PWR (2 ass.)	BWR (6 bdls)
normal test	surface of the outer shell	75	76
	surface of the inner shell	135	162
	gasket	72	80
	fuel cladding	211	229
accident test	surface of the outer shell	818	818
	surface of the inner shell	321	319
	gasket	233	233
	fuel cladding	247	260

4.3 Containment Analysis

The leakage of nuclear materials from the containment has been analyzed under the following assumptions.

① The nuclides are treated as powder.

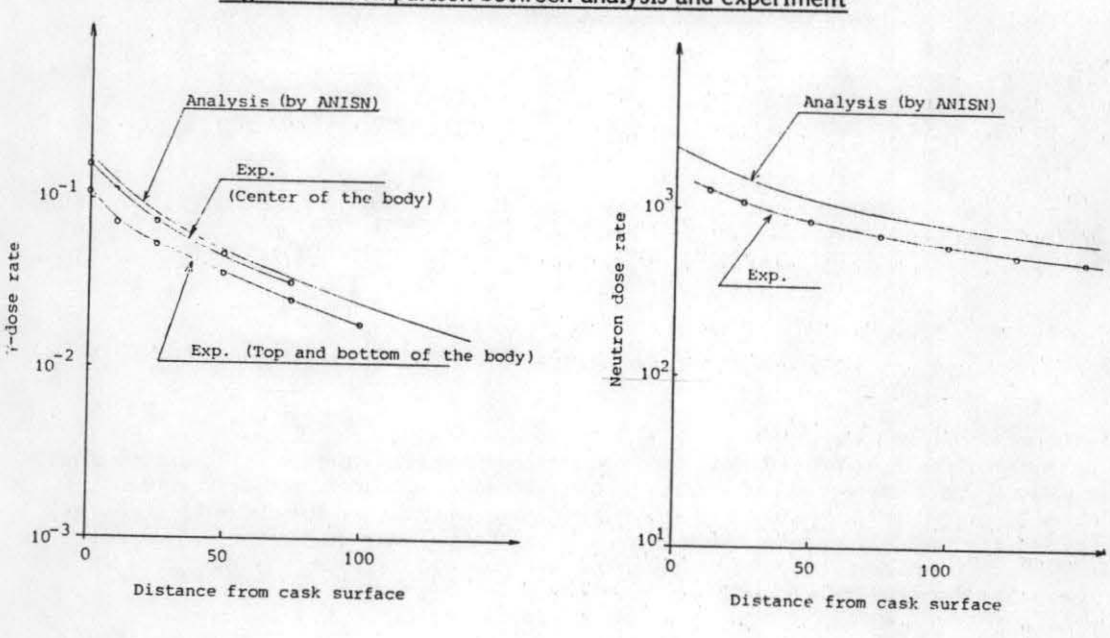
② The primary containment boundary (fuel cladding) and secondary containment boundary (gaskets) are intact.

The leakage of the nuclides is evaluated in accordance with NUREG CR/1302 (2). It has been confirmed leakage of the activities does not exceed the allowable value in normal and accident test conditions ($A_2 \times 10^{-6}$ Ci/hr in normal test condition and A_2 Ci/week in accident test condition).

4.4 Shielding Analysis

The shileding analysis has been performed by ANISN code. The maximum radiation dose rate is 7.8 mrem/hr for the package containing BWR fuel at 1m away from the container surface under normal condition. The maximum dose for both packages either containing PWR or BWR fuels is approx. 500 mrem/h under accident condition assuming that the container deforms and the shock absorber and neutron shielding are neglected. And in order to clarify the adequacy of analysis method the shielding test has been performed as described in 5.2.2, then the analysis method is confirmed to be adequate by comparing the results of the test and the analysis data of ANISN calculation code as shown in Fig. 4.4-1.

Fig. 4.4-1 Comparison between analysis and experiment



4.5 Criticality Analysis

The packages have been analyzed by the KENO-IV Monte Carlo Code and 16 groups cross section library of Hansen-Roach under normal, and accident conditions. The criticality analysis are carried out considering following conditions.

① The optimum water density between the array of the packages.

② Reduction of spaces between packages and those of fuels.

③ B_4C contained silicon rubber is neglected under accident conditions.

The maximum effective multiplication factor ($K_{eff}+3\sigma$) is obtained when the packages contain BWR fuel under accident conditions and the value is 0.90.
It means that in any cases, the value of $K_{eff}+3\sigma$ is less than 0.95 and it has been confirmed that the packages are designed to retain subcriticality even if infinite number of packages are arranged arbitrarily.

5. Outline of the Tests and Their Results

Regarding with the development of the container the following tests have been performed in order to confirm the validity of the safety analysis and the compliance as type B package requirements.

① Strength tests and thermal tests using scale models and slice models respectively

② Handling, transport and shielding tests by prototype models.

③ Strength, thermal and immersion test by prototype models.

And the purpose of these tests are shown with sequence as follows :

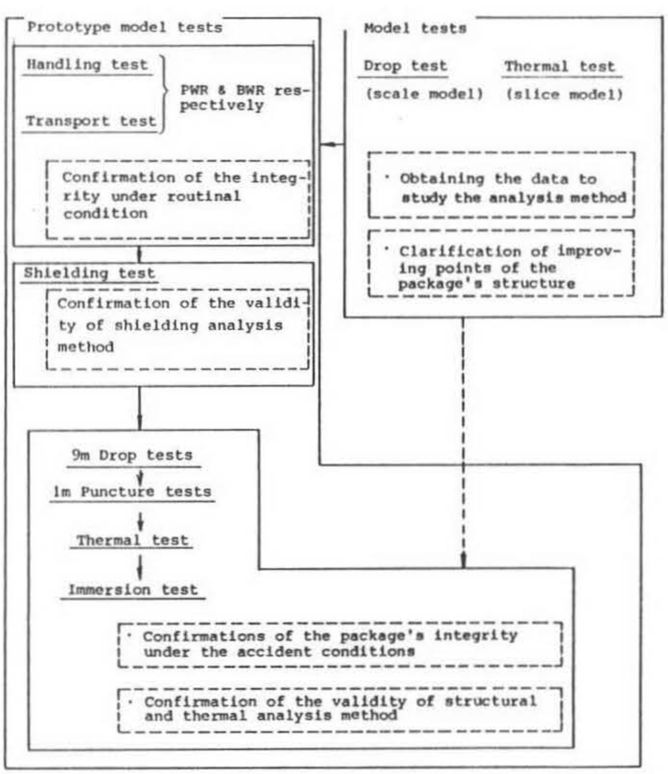


Fig. 5-1 Test sequence and its purpose

Symbol

↓ : sequence

⌜⌟ : purpose

5.1 Specimen

5.1.1 1/3 Scale Model

Two 1/3 scale models were manufactured in order to provide the strength tests (9m free drop and 1m puncture tests). The main specifications are shown in Table 5.1.1-1.

Table 5.1.1-1 Specification of 1/3 scale model

Classification	Actual Design	1/3 Scale Model	Scale Factor
External Dimension OD (m) Full Length (m)	 1.3 5.5	 0.43 1.83	 3.0 3.0
Weight (MT)	9.1	0.36	25.3
Material	Stainless steel	Stainless steel	-

5.1.2 Slice Model

The Slice models were manufactured to simulate the lid part and the middle part of the actual package to verify the thermal analytical method.

5.1.3 Prototype Model

Two prototype models were manufactured. The one is a model for containing 2 PWR fuel assemblies and the other is a model for containing 6 BWR fuel bundles. The above two prototype models have the same dimensions and weights and using the same materials as the actual design.

The outlook of the prototype model is shown in Fig. 5.1.3-1.

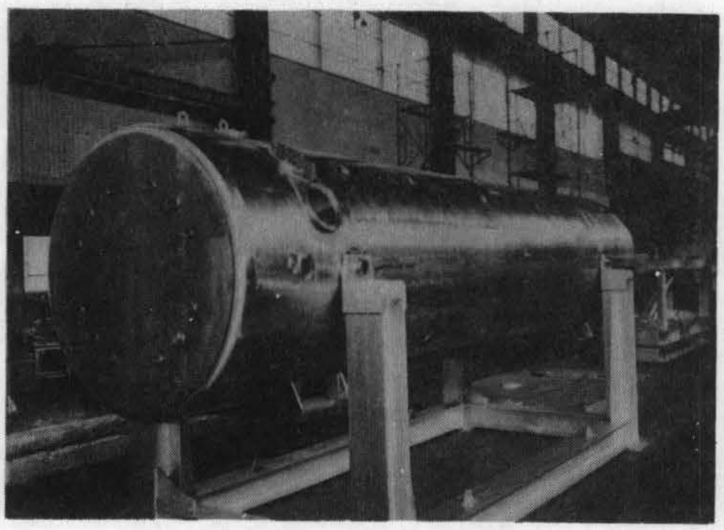


Fig. 5.1.3-1 Prototype-model

5.2 Tests Results

5.2.1 Results on the Drop Test Using the 1/3 Scale Models

By using two 1/3 scale models, 9m drop/1m puncture tests have been carried out by the attitude of ① Vertical, ② Horizontal, ③ Corner, and ④ Oblique (an angle of 45°) respectively.

The data obtained by the tests provides the validity of the analysis method (described in 4.1). The containment integrity can be maintained in any drop attitudes of the scale models. (Fig. 5.2.1)

5.2.2 Results on the Thermal Tests Using the Slice Models

The thermal tests have been performed at 800°C for 30 minutes in the furnace. The analytical method is confirmed to be adequate by the comparison between the analysis and the tests. (Fig. 5.2.2)

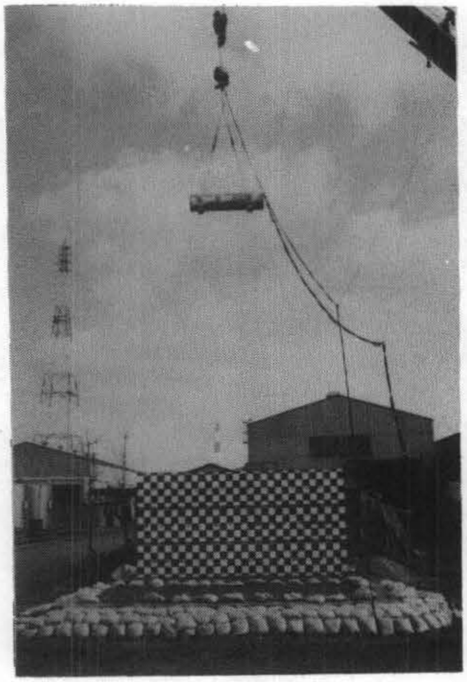


Fig. 5.2.1 Drop test by 1/3 scale model

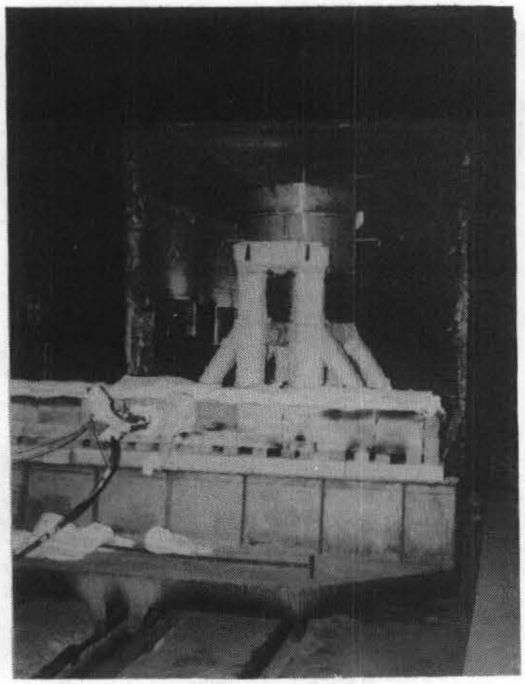


Fig. 5.2.2 Thermal test by slice model

5.2.3 Results on the Handling, Transport and Shielding Tests

The handling tests are conducted using the prototype models containing dummy fuel assemblies either BWR or PWR to confirm the handling performance for the packaging and fuel supporting system. As the results, their handling performance is satisfactory for activities in the fuel manufacturing facility and the nuclear power stations.
Then the transport tests are carried out to simulate actual transport by the prototype package. The results of these tests show that the prototype packages are intact with no damage.
And shielding test has been performed by using Cf^{252} and Co^{60} source respectively. The results of the shielding tests are satisfactory; as already shown in Fig. 4.4-1.

5.2.4 Results on the Drop, Thermal and Immersion Tests

The prototype model for BWR has been subjected to 9m drop and 1m puncture tests (the attitude is vertical), thermal test (800°C, 30 min) and immersion test (a depth of 15m, 8 hrs) and the one for PWR has been subjected to 9m drop and 1m puncture tests (the attitude is horizontal).

(1) 9m Drop and 1m Puncture Tests (Fig. 5.2.4-1)
From the tests, the package's integrity and the validity of the analysis methods have been demonstrated as described in 4.1.

(2) Thermal test (Fig. 5.2.4-2)
The thermal test have been performed at 800°C for 30 minutes. The integrity of the containment system and the validity of thermal analytical methods have been satisfactory demonstrated by test results.

(3) Immersion test
After the thermal test, the prototype model have been subjected to immersion test sequentially. The integrity of the containment systems of the package is well confirmed by test result.

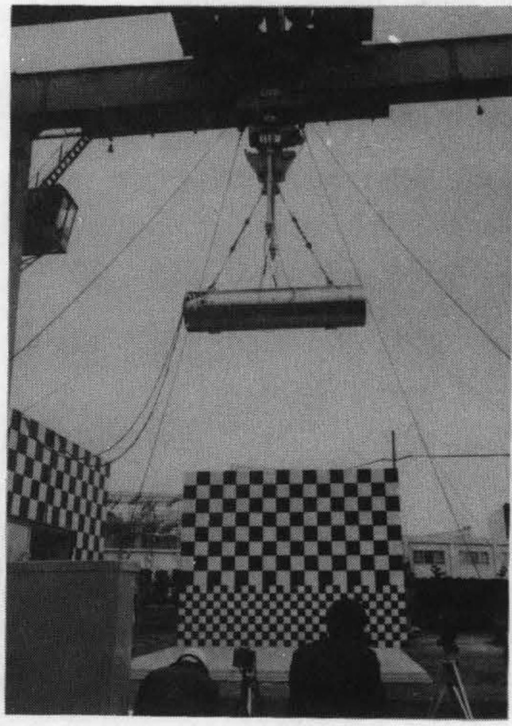


Fig. 5.2.4-1 Drop test

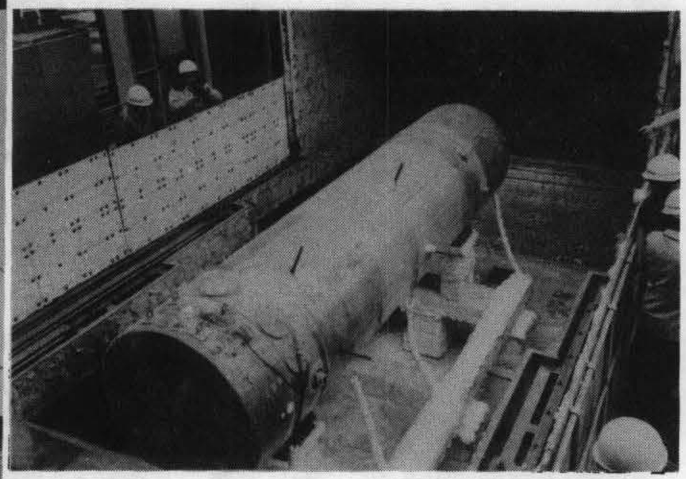


Fig. 5.2.4-2 Thermal test

6. Conclusion

The shipping package for MOX fresh fuel have been developed and confirmed to keep their integrity during handling and transport the MOX fuels with safe and economical way.
And these packages have been demonstrated to meet all the requirements of the Regulations.

7. Acknowledgement

The transport tests were performed by the test facilities of JARI (Japan Automobile Research Institute). And the shielding test and the sequential tests (9m drop, 1m puncture, etc.) were performed by the test facilities of CRIEPI (Central Research Institute of Electric Power Industry). The authors acknowledge specially the members concerned of the tests in these two institutes.

8. Reference

(1) Regulations for the Safe Transport of Radioactive Materials 1985 Edition, IAEA

(2) NUREG/CR1302 Study of Plutonium Oxide Powder Emissions from Simulated Container Leaks.