Development of the Nuclear Ship MUTSU Spent Fuel Shipping Cask

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INTRODUCTION

After the planned trial voyage (4700 MWD/MTU) of the nuclear ship MUTSU in 1990, her spent fuel assemblies, initially made of two types of enriched UO_2 (3.2wt% and 4.4wt%), will be transferred to the reprocessing plant soon after cooling down in the ship reactor for more than one year. For transportation, the MUTSU spent fuel shipping casks will be used. Prior to transportation to the reprocessing plant, the cooled spent fuel assemblies will be removed from the reactor to the shipping casks and housed at the spent fuel storage facility on site. In designing the MUTSU spent fuel shipping cask, considerations were given to make the leak-tightness and integrity of the cask confirmable during storage. The development of the cask and the storage function demonstration test were performed by Japan Atomic Energy Research Institute (JAERI) and Mitsubishi Heavy Industries, Ltd. (MHI) according to the schedule shown in Table 1. One prototype cask for the storage demonstration test and licensed thirty-five casks were manufactured between 1987 and 1988.

Table 1. Development Schedule of the Nuclear Ship MUTSU Spent Fuel Shipping Cask

F.Y.	1985	1986	1987	1988
Spent fuel shipping cask		Conceptual design Basic design	Basic design Detailed design/Licensing Manufacture	Manufacture Installation
Storage function demonstration test (prototype cask)			Design Manufacture (prototype)	Storage demonstration test

DESIGN CONDITIONS

The design conditions of the cask are listed below:

- Capability of handling on and inside of nuclear ship MUTSU. The basic flow diagram of the handling is shown in Fig. 1.
- Design as a type B(U) fissile class 1 package.
- Long term containment performance and function to confirm the containment integrity during storage on condition that the casks are used as part of the storage facillity.
- Containing the spent fuel which is specified in Table 2.

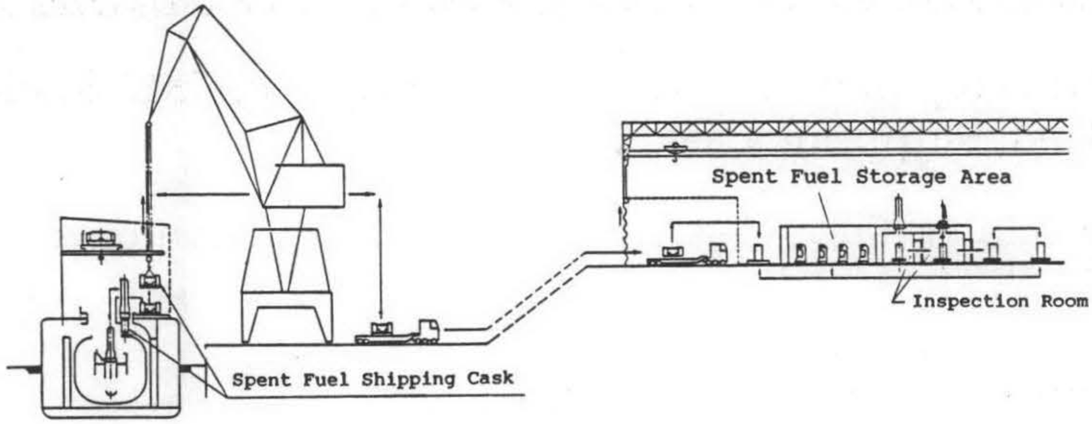

Fig. 1 Handling flow diagram of the cask

Table 2. Schematic specification of the nuclear ship MUTSU spent fuel shipping cask

Name of package	MUT-87Y-15T	
Type of package	B(U) type fissile class 1	
Max. weight of package (metric ton)	15	
Cask dimension (mm) - Length with impact limiters - Diameter with impact limiters	 2990 1500	
Material of cask - Cask body - Fuel basket - Impact limiter (+ Cover)	 Stainless steel Alminum + Stainless steel Fir-plywood (+ Stainless steel)	
Specification of fuel assy. to be housed	Type 1	Type 2
- Nuclear fuel material	UO_2	UO_2
- Number of assemblies	1	1
- Weight of UO_2 (kg)	∿87	∿87
- Initial enrichment to analysis (wt%)	∿3.3	∿4.5
- Burnup (MWD/MTU)	4,700	4,700
- Cooling time (day)	365	365
- Heat generation (W)	212	207
- Radioactivity (PBq (Ci))	1.92 (5.2×10^4)	1.89 (5.1×10^4)

OUTLINE OF STRUCTURE

Characteristics

(1) The cask is able to store the spent fuel as part of the storage facility through taking the following into consideration :

- To maintain the integrity of the spent fuel under storage, the cask cavity is filled with helium gas.
- To maintain the containment integrity during the long term storage, a double containment structure which comprises the primary lid and secondary lid each having a couple of metal and elastomer O-rings is employed.
- The pressures of helium gas within the cavity and between the lids are continuously monitored by pressure transducers to confirm the containment integrity of the cask.

(2) The cask body is very simple, being composed of a single layer structure made of forged stainless steel.

Specification

A schematic drawing of the cask is shown in Fig. 2. Fig. 3 shows the outlook of the cask, and its specification is shown in Table 2. The casks are handled vertically when they are lifted or placed for loading/unloading or for storage of the fuel assemblies, and in the case of transportation, they are laid horizontally.

Cask Structure

A single fuel assembly is accommodated in the cask because of the handling limitation inside the nuclear ship MUTSU. The cask consists of the cask body, primary lid, secondary lid, fuel basket and impact limiters, as shown in Fig. 2.

(1) Cask body

The cask body is made of forged stainless steel of 345mm thickness determined for gamma and neutron shielding. The cask body composes a containment boundary in combination with the primary lid.

Vent valves and drain valves are installed on the top and bottom sides of the cask body respectively, which are used for vacuum drying, helium gas filling, draining, etc. in conditioning the cask cavity. When the cask is used for storage, two pressure transducers are installed on the top side of the cask body to monitor the gas pressures in the cavity and in the space between the primary and secondary lids.

(2) Primary lid and secondary lid

The primary lid is made of stainless steel, having dimensions of 648mm O.D. and 275 mm thickness, while the secondary lid is also made of stainless steel, having dimensions of 860mm O.D. and 70mm thickness. Each of them is fixed on the cask body by means of 24 bolts. The coupling parts of the lids and the cask body are each sealed by double O-rings (inside : metal O-ring, outside : elastomer O-ring).

(3) Fuel basket

The fuel basket has a dual function to support the fuel assembly and to transmit the heat from the fuel to the cask body during transport and storage.

As for the fuel basket materials, aluminum alloy is used for the trunk part, and stainless steel is used for both the upper and lower ends.

(4) Impact limiters

In order to protect the cask body and fuel assembly, etc. against a mechanical impact caused in a hypothetical accident, impact limiters which are made of fir- plywood covered with stainless steel plates are fixed on the upper and lower ends of the cask during transport.

SAFETY ANALYSIS

The cask must be used safely for transport and storage of spent fuel, so safety analyses were carried out on the loaded cask as a transport package and as a component of the storage facility.

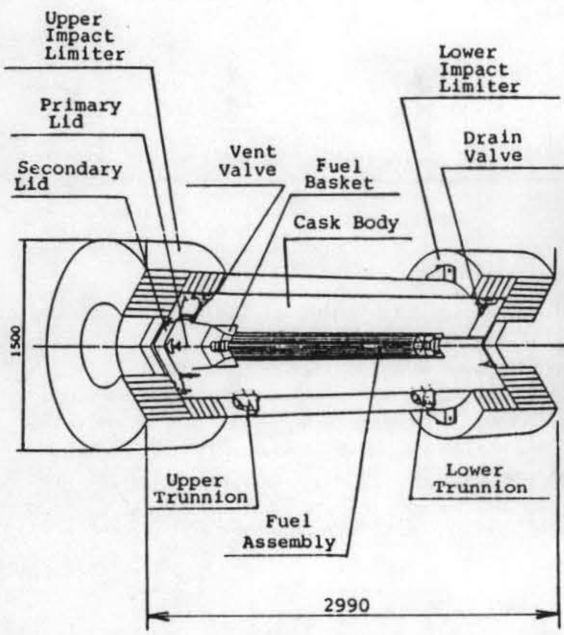

Fig. 2 Schematic drawing of the nuclear ship MUTSU spent fuel shipping cask

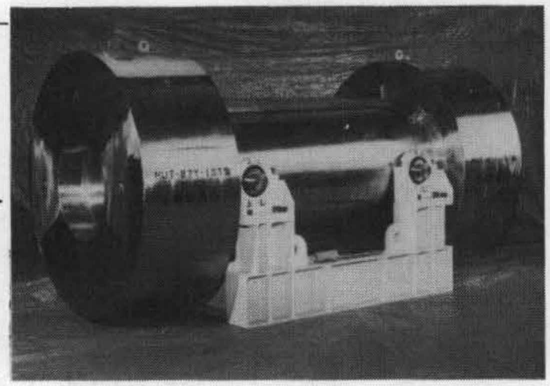

Fig. 3 Outlook of the cask

Structural Analysis

(1)Transport

- The structural analysis of the cask was carried out in consideration of "Design criteria for the structural analysis of shipping cask containmant vessels" (USNRC R.G. 7.6).
  A strain limit was employed in the design criteria for the primary lid to ensure the containment integrity in the accident conditions.
- The analysis of the acceleration of the package and the deformation of the impact limiters in the drop test was carried out by CASH-II code (Asada et al. 1988). For the vertical drop test, the analysis by DYNA-2D code was also carried out so as to verify the analysis results by CASH-II code.
- The static analysis of the cask body was carried out using MARC code which is an FEM (Finite element method) code.
- Assuming an accident during the transport by sea, the analysis of the buckling and stress of the cask body was carried out to evaluate the integrity of the package in case of a 3000m immersion.

(2) Storage

The analysis of pressure-resisting strength was carried out by MARC code on the basis of the criteria for a class 1 component specified in ASME Code Sec. III, Subsec. NB.

Thermal Analysis

(1) Transport

The thermal analysis using MARC code was carried out to confirm that the surface temperature of the cask under transport would not exceed 82°C, and the cask would be able to withstand a fire condition with a temperature of 800°C for a period of 30 minutes. The results are shown in Table 3.

(2) Storage

It was verified through analysis using MARC code that the casks had adequate thermal characteristics under the storage of spent fuel (assuming the ambient temperature at 48°C) in the storage facility, and the temperature of the fuel rod cladding would not exceed 400°C. The results are shown in Table 4.

Table 3. Results of thermal analysis (transport)

Position	Maximum temp. in normal conditions (° C)	Maximum temp.in thermal test conditions (° C)	Hours after fire occurance (Hr)
Fuel rod cladding	143	145	3.3
Inner surface of fuel basket	73.2	142	3.3
Inner surface of cask body	72.5	148	3.3
Outer surface of cask body	70.0	389	0.5
O-ring part of primary lid	69.2	103	2.9

Table 4. Results of thermal analysis (storage)

Position	Maximum temperature (°C)
Fuel rod cladding	129
Inner surface of cask body	60.7
Outer surface of cask body	58.5
O-ring part of primary lid	56.0

Sealing Analysis (Transport/Storage)

- The cask cavity is evacuated to a negative pressure before transport. So it was confirmed that the cavity was maintained in a negative pressure under the transport accident conditions, and radioactive materials did not leak.
- On the safe side, it was evaluated that even if the fuel rod cladding should be fractured, the leakage of radioactive materials would not exceed the regulatory value.

Sheilding Analysis

(1) Transport

The neutron and gamma source intensities were calculated by the isotope generation and depletion code of ORIGEN, and the effective multiplication factor of the neutron source was calculated by the multi-group Monte Carlo method code of KENO-IV.

Neutron and gamma dose rates were calculated by the one-dimensional transport equation code of ANISN and two-dimensional transport code of DOT-3.5, respectively. It was confirmed that the total dose rate was not grater than 2 mSv/h (200 mrem/h) on the cask surface and 0.1 mSv/h (10 mrem/h) at 1m from the surface under the normal conditions.

(2) Storage

The shielding design of the storage facility was performed on the assumption that the dose rate of the cask was 0.25 mSv/h (25 mrem/h) or less on the surface and 0.08 mSv/h (8 mrem/h) or less at 1m from the surface from the analysis results on the package above.

Criticality Analysis (Transport/Storage)

The criticality analysis was carried out by KENO-IV code. It was confirmed that the effective multiplication factor (Keff+3σ) was 0.843 (<0.95) on the assumption that an infinite number of casks whose cavities were kept in the optimum water density condition of fission congregated arbitrarily.

OUTLINE OF THE STORAGE FACILITY

The spent fuel storage area is part of the spent fuel storage facility (in a reinforced concrete building having dimentions of appoximately 22m $\times$ 67m $\times$ 23m (H)), as shown in Fig.4, Fig.5 and Fig.6, which has been recently installed on the nuclear reactor facility site of the nuclear ship MUTSU. In the facility there are a spent fuel storage area, fresh fuel storage area, etc.

In the spent fuel storage area, there are 35 cask skids, on which the casks have been already placed in a vertical array as shown in Fig.7.

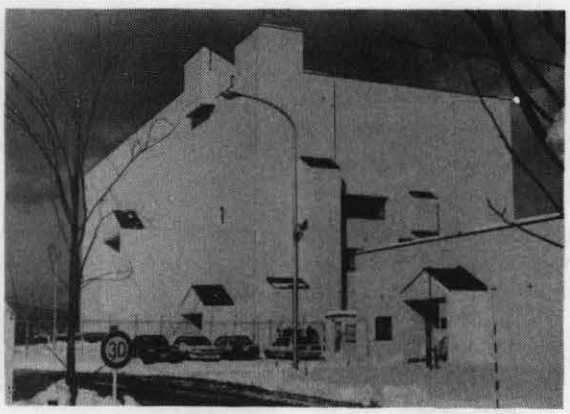

Fig. 4 Outlook of the fuel and waste treatment facility

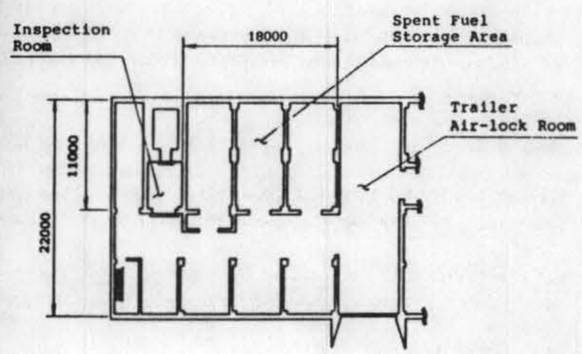

Fig. 5 Ground plan of fuel and waste treatment facility

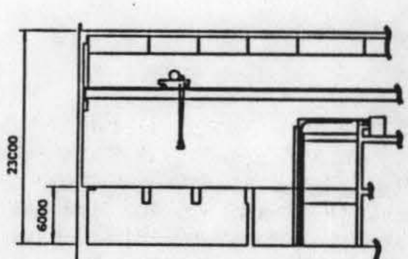

Fig. 6 Cross section of fuel and waste treatment facility

Fig. 7 Casks in the storage facility

STORAGE FUNCTION DEMONSTRATION TEST

Outline

Since the casks are used as storage components of the spent fuel storage facility, the storage function demonstrationtest was carried out during the period from late 1987 to mid 1988. Its purposes were to demonstrate the function of the casks as part of the storage facility and to obtain the basic data to establish the management criteria for the casks and the spent fuel assemblies during storage. A test building, where the ambient temperature was controllable to simulate that of the actual spent fuel storage area, was built in MHI Takasago R & D Center. A thermal performance test, a pressure transducer function test and a handling test were performed in the building, using the prototype cask and a dummy fuel assembly. The protytype cask is shown in Fig. 8, and the test building is shown in Fig. 9.

Thermal Performance Test

The prototype cask was installed in the environment of the test building where the temperature of the storage area was simulated. Under this condition, the thermal performance test was performed taking the environmental temperature and heat generation of the dummy fuel assembly as parameters, and the temperatures at various parts of the prototype cask and the dummy fuel assembly were measured. Some test results are exemplified in Fig. 10 and Fig. 11. The relation between the temperature of the cask surface and the maximum temperature of fuel assembly was confirmed.

Pressure Transducer Function Test

The tests simulating the status where a fracture of fuel rod cladding or containment boundary of the cask occurred and thereby the cavity pressure changed were performed. The response of the pressure transducers were confirmed to be adequate.

Handling Test

(1) The handling test on the cask (opening/closing the lids, leak rate measurment and helium gas filling, etc.) was performed, and the operation time and procedures required for the routine cask handling were clarified.

(2) The vacuum drying test on the cask was performed, and the operation time and procedures required for the vacuum drying were clarified. An example of the test results is shown in Fig. 12.

MANUFACTURING

35 casks in total were manufactured for practical use. Under the applications made by JAERI, the inspections of the packagings were carried out by the Science and Technology Agency (STA) and the Ministry of Transport (MOT), and the inspections of the spent fuel storage facility were carried out by STA. After the manufacture of all the casks was accomplished successfully in June 1988, the registrations of the packagings were issued by STA and MOT in August 1988. The approval as a nuclear facility was obtained from STA in October 1988.
Fig. 7 shows the casks in the storage facility.

CONCLUSIONS

JAERI and MHI developed the nuclear ship MUTSU spent fuel shipping cask between 1985 and 1988 through the considerations and analyses to satisfy its transport requirements and planned storage functions. In the latter part of that period, the storage function demonstration test was carried out using the prototype cask on its performances in heat dissipation, containment integrity, cavity condition monitoring as well as various handling aspects, and thereby the expected functions and operabilities of the casks as part of the spent fuel storage facility for the nuclear ship MUTSU were verified.
The manufacture of 35 units of the developed casks was already completed, and the registrations for their practical use have been issued by STA and MOT, while the approval for the spent fuel storage facility has been obtained from STA.
The casks will be used effectively for the safe storage and transport of the spent fuel of the nuclear ship MUTSU.

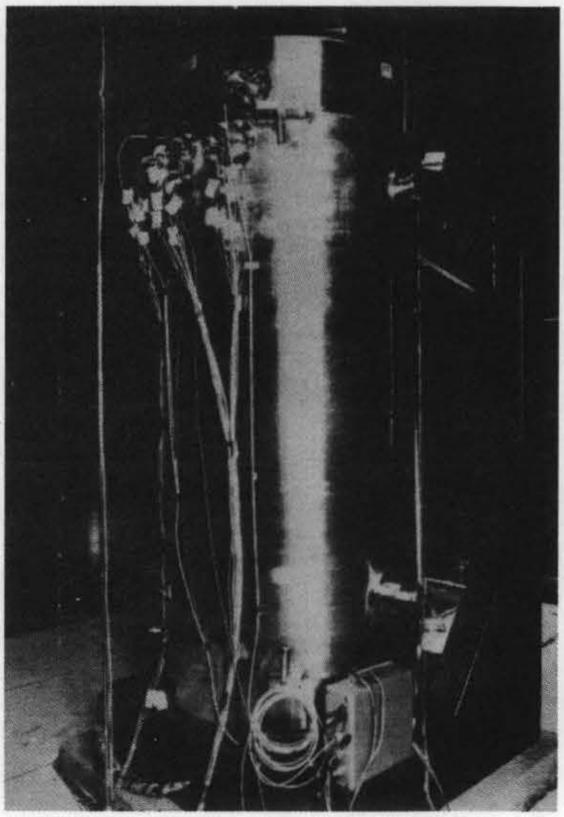

Fig. 8 Prototype cask

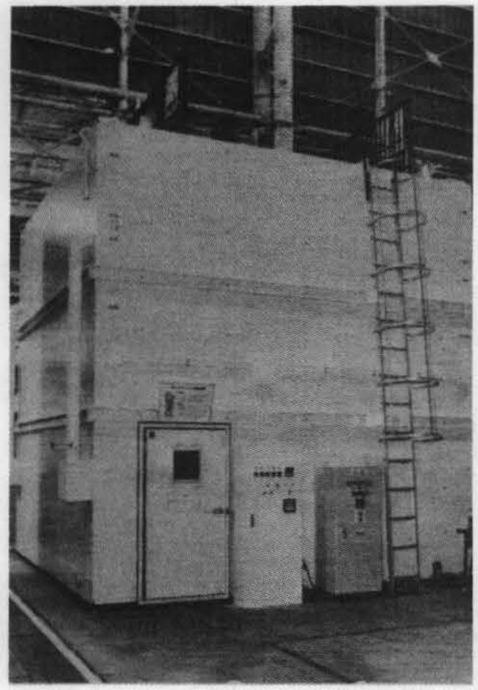

Fig. 9 Test building

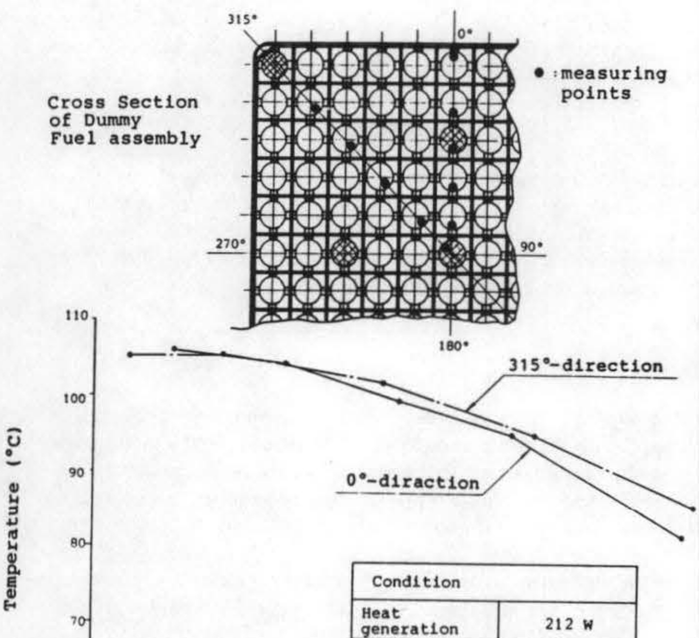

Fig. 10 Radial temperature distribution of dummy fuel assembly (Example)

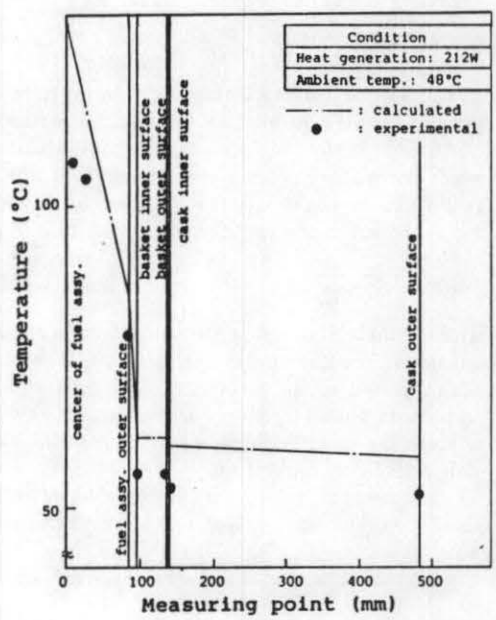

Fig. 11 Radial temperature distribution in steady state (Example)

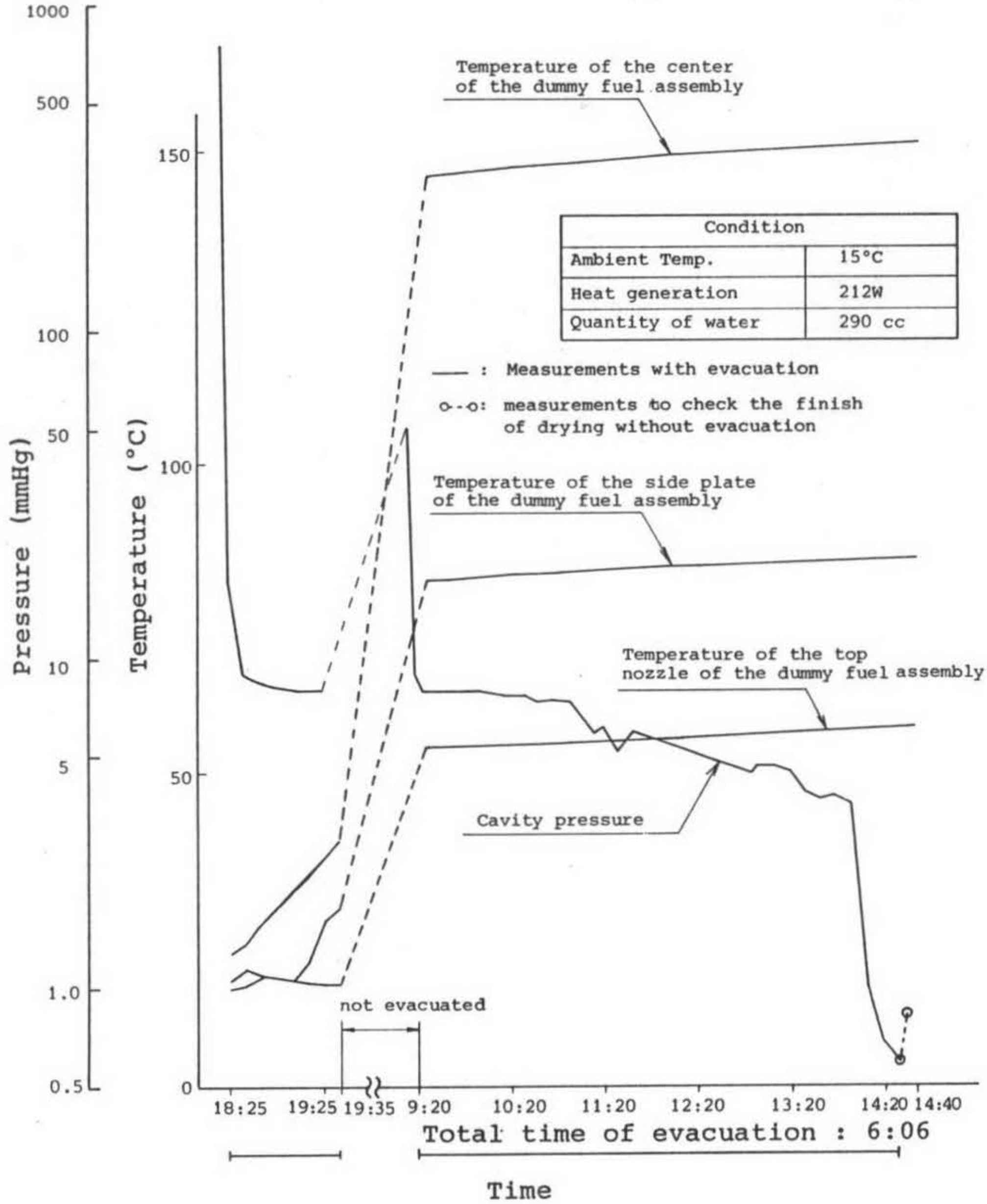

Fig. 12 Results of the vacuum drying test (Example)

REFERENCE

A.Asada, M.Ohashi, et al., "Development of Simplified Analysis Codes for 9-M Drop and 1-M Puncture Tests for a Radioactive Material Transport Cask", Waste Management '88(1988).