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INSTITUTE OF NUCLEAR MATERIALS MANAGEMENT

On the Annual Meeting, Workshops, and Student Activities

By Cathy D. Key
INMM President

Annual Meeting

The 46th Annual Meeting of the INMM will be held July 10–14, 2005, at the JW Marriott Desert Ridge in Phoenix, Arizona, U.S.A. We held the 2003 meeting at this same spot and it was very well attended, very successful, and enjoyed by all. We look forward to another successful annual meeting this year. All activities to set up this meeting are well under way and on track. Let me review a few important dates for those of you that will be presenting papers.

April 8	E-Mail Notification of Acceptance
April 15	Speakers Manual Available
June 9	Speaker Registration Due
June 10	Final Papers Due
June 10	Changes and Withdrawals Due

As of this writing more than 300 abstracts for presentations have been submitted to the official INMM abstract database. I don't have to remind anyone that the success of the INMM Annual Meeting is based upon the emphasis on presentations. We appreciate your continued interest and involvement in assuring that the meeting is successful. The annual meeting provides attendees with a professional forum for the exchange of the latest technical information in nuclear materials management. It is important that we all take advantage of this forum and participate. On March 8, Charles Pietri once again led the Technical Program Committee through its annual exercise of massaging all of the submitted abstracts into the sessions that make up the annual meeting.

Be sure to register for the 46th INMM Annual Meeting online at www.inmm.org.

Upcoming and Recent Seminars and Workshops

Our Nonproliferation and Arms Control Technical Division, along with our Northeast Chapter sponsored a workshop, "A New Role for the United Nations Security Council: Criminalizing WMD Proliferation: the Impact of U.N. Security Council Resolution 1540," March 15. I hope that many of you were able to attend this workshop.

The Physical Protection Technical Division will hold a workshop in Augusta, Georgia, U.S.A. April 25–28, 2005, on "Safeguards Security System Effectiveness." I am sure that this workshop will be successful also.

May 16–20, 2005, the Third Russian International Conference on Nuclear Material Protection, Control, and Accounting will be held in Obninsk, Russia. This conference is listed on the INMM Web site (www.inmm.org) and gives you the opportunity to download all necessary information concerning the conference including registration. We wish our Russian colleagues success in this workshop, which is co-sponsored by the INMM.

Before the annual meeting there will be a Workshop on Developing Physical Protection Specialists July 7–8 at the JW Marriott Desert Ridge. You can view the details of this workshop on the INMM Web site (www.inmm.org). Each year, we hold topical workshops that have high participation. This highlights the ever-present importance of everyone working together, sharing information regularly, and keeping abreast of everything going on in our professions.

Student Activities

Student activities remain a high priority in our organization. The annual meeting will again provide an award for the best student paper presentations. We look forward to continued student participation. Also, there is work in progress to develop the first INMM Student Chapter. This chapter will be at Texas A&M University under the official advisement of William Charlton. We appreciate the effort shown thus far in the development of this chapter. We are very excited for (and about) this chapter. Students are the future for the continued excellence of our profession.

INMM President Cathy D. Key may be reached by e-mail at cathykey@key-co.com.

Technical Editor's Note

By Dennis Mangan
Technical Editor

Full-Scale Drop Testing of the CONSTOR®, V/TC Package Program, and Preliminary Results

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Introduction

The CONSTOR® steel-sandwich cask was developed as a cost-effective cask design by using conventional mechanical engineering technologies and commonly available materials. The CONSTOR® consists of a cask body with an outer and a steel inner liner. At the upper end, the liners are welded to a ring made of forged steel. The space between the two liners is filled with heavy concrete called CONSTORIT for additional gamma and neutron shielding. The CONSTORIT consists of an iron aggregate frame and hardened cement paste. The liners of the cask and the forged head ring form the basis for the structural integrity; the CONSTORIT plays only a subordinate part in case of accident loads.

The CONSTOR® casks are for the storage and transport of light-water reactor spent fuel. The CONSTOR® concept fulfills both the internationally valid International Atomic Energy Agency (IAEA) criteria for transportation and the requirements for transportation and long-term intermediate storage in the United States and various European countries.

Since the beginning of the development of the CONSTOR® design in the early 1990s, two drop test series with half-scale models have been performed. In 1997, a drop test program containing HAC (hypothetical accident condition) free drops and HAC puncture tests[1] was performed with the CONSTOR® VB-1, a 1:2 model of a CONSTOR® RBMK 1500 that was designed and licensed for the storage of RBMK fuel in Lithuania (there the heavy concrete has an additional baryte aggregate). To date, sixty CONSTOR® casks of this type have been delivered to the Ignalina Nuclear Power Plant. They are loaded and stored in open air onsite.[2] In 2002, another drop test program with an advanced CONSTOR® cask design (CONSTOR® VB2) was performed.[3] The geometry was the same as for the CONSTOR® VB1. To improve the heat removal properties in comparison with the CONSTOR® VB1, heat conducting elements were arranged inside the CONSTORIT. The name for the new CONSTOR® cask series is CONSTOR® V.

The results of the CONSTOR® VB-2 drop test program showed that the integrity and leak tightness of the liner welds were maintained over the test series. However, the puncture test proved to be the most demanding condition for the cask integrity. To enhance the analytical mechanical safety margins during transport accidents, the enlargement of the liner thickness was discussed. As a result, the development of the puncture-resistant jacket started. Instead of enlarging the cask wall thickness for the transport, the intended purpose was achieved by an additional steel liner. It provided the opportunity to separate requirements for the storage and the transport applications.

By this means, handling procedures at the facilities did not need to be complicated because of the enhanced cask weight. Also, the economics of the transport and storage system are less affected because only a small number of puncture-resistant jackets are needed. The design of the puncture-resistant jacket was realized for the first time for the CONSTOR® V/TC.

A full-scale model of a CONSTOR® V for BWR inventory has been manufactured and was presented as safety demonstration by the Bundesanstalt für Materialforschung und -prüfung (BAM), in an HAC free drop test on the drop test facility at Horstwalde near Berlin during the PATRAM 2004 Symposium. The CONSTOR® V/TC package consists of the cask with a dummy for the basket and the fuel assemblies (FAs), a puncture-resistant jacket and two impact limiters.

The Full-Scale CONSTOR® V/TC Package

The CONSTOR® V/TC represents a prototype of a CONSTOR® V/69 — designed for sixty-nine BWR fuel assemblies and is very similar to the CONSTOR® V/32, which is designed for 32 PWR fuel assemblies. The CONSTOR® V/TC also represents the basic CONSTOR® design. It consists of a cask body with an outer and a steel inner liner. At the upper end, the liners are welded to a ring of forged steel. The space between the two liners is filled with CONSTORIT. In addition to the basic CONSTOR® design, it has the features of the advanced CONSTOR® design for heat loads up to 30 kW, which was firstly realized with the building of the CONSTOR® VB-2; Copper heat conducting elements (see Figure 1), and CONSTORIT (see Figure 2). The completed cask body is shown in Figure 3.

Figure 1. Copper heat conducting elements between the liners of the cask body

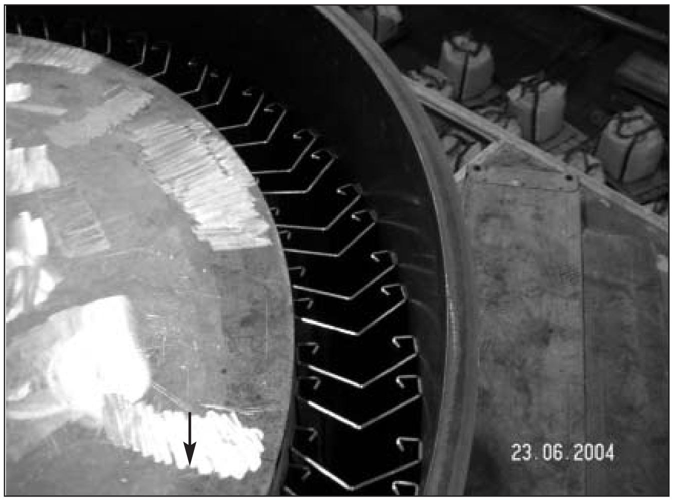


Figure 3. CONSTOR® V/TC general view and FAs

Figure 2. CONSTORIT specimen between the liners of the cask body

Figure 4. CONSTOR® V/TC dummy basket

The lid system comprises a steel shielding lid resting on an inclined ring surface in the head ring and fixed in the axial direction by means of form-fit retaining elements in a groove within the head ring, as well as a closure lid with bolts and sealed with two metal O-rings. Thus, the transport containment is constituted by the inner liner, the head rings and the closure lid with the two metal O-rings.

The mass of the basket and the inventory are provided in the form of a dummy (see Figure 4) consisting of axially arranged steel disk elements, with torsion locks and spacers. The torsion lock device ensures that the instrumentation for strain and acceleration measurements cannot be damaged during the tests. The basket dummy represents the mass of the fuel elements and the basket as well as the load applied to the cask body during the HAC free drop and the HAC puncture tests. The total weight of the dummy (~36 metric tons) corresponds to the weight of the basket with fuel elements of the CONSTOR® V/69.

Between the upper end of the inventory dummy and the shielding lid, the largest possible axial distance is selected. This ensures that in case of a secondary impact (drop position onto the lid system or onto the bottom side) the largest possible load is applied to the shielding lid or the bottom of the cask.

Octagonal impact limiters were used for the tests. Bottom and lid impact limiters consists of several layers of encapsulated wood. To resist the puncture drop test, a steel plate is integrated into the impact limiter structure covering the closure lid. Additionally, a steel ring is welded to this plate, to protect the cask during the puncture test at its lid and bottom ends in

Figure 5. CONSTOR® V/TC transport package at the GNS Assembly Plant in Mülheim


Figure 6. Puncture-resistant jacket (tongue)

horizontal drop positions. The impact limiters are attached to the puncture-resistant jacket by bolts (see Figure 5). The puncture-resistant jacket itself consist of a thick steel shell made of two separate parts with form-fit profiles in the contact area (see figures 6 and 7). In 2004, the prototype of the CONSTOR® V/TC was manufactured. In tables 1 and 2, the geometrical data and the masses of the CONSTOR® V/TC package and its components are shown. The cask body features two passages through the cask wall as cable conduits. The measuring cables of redundant strain gauges can thus be fed out of the cask at different points.

On the inside of the cask, a U-profile welded across the entire axial shaft length of the inner liner is provided to prevent any torsion. This U-profile is also used as a cable conduit. The above-mentioned passages in the cask wall are on the same level as the U-profile in order to provide an easier cable routing.

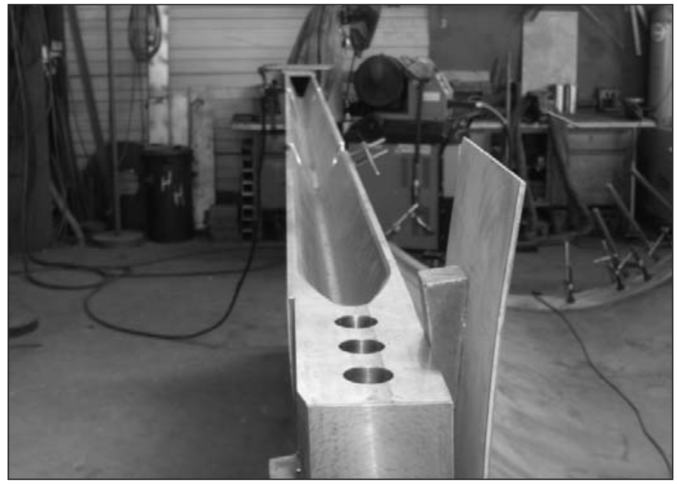


Figure 7. Puncture-resistant jacket (groove)

Surfaces on the external cask side or on the inside of the impact limiters are, as much as required, fitted with cable conduits in order to prevent measuring cables from shearing off. The same measures are taken on the inner liner bottom in order to route the measuring cables to the axial conduit. At the inventory dummy, however, recesses are provided to protect the cables at the inner liner of the cask.

Test Objectives

According to the transport regulations, the structural evaluation of a transportation package may be performed by analysis, test, or a combination of both. The structural evaluation of the CONSTOR® transportation package is performed by analysis. However, the demonstration test (horizontal HAC free drop) was performed to show the adequacy of the analytical tools and assumptions used for the structural analysis. This will be achieved through comparisons between the measured results from the drop test and the pre-test predictions made by analysis.

Pre-Test Predictions

The pre-test predictions for the HAC free drop test with the CONSTOR® V/TC package were performed using the same analytical and numerical tools and assumptions that will be used for the safety analysis of the CONSTOR® V/69 and CONSTOR® V/32 package designs.

The pre-test predictions were based on the anticipated test conditions, considering the real specific mass and material properties of the test package. Specific items include the dynamic acceleration time-history response of the package at the instrumented locations, dynamic strain response at specific locations on the package, and overall damage predictions (i.e., impact limiter deformations and permanent deformation of package components).

Table 1. Geometrical data of the CONSTOR® V/TC package and its components

Package length with impact limiters	7 445 mm / 293 in
Outer Diameter of the cask	2 332 mm / 92 in
Outer Diameter of the impact limiters	3 510 mm / 138 in
Outer Diameter of the puncture-resistant jacket	2 592 mm / 102 in

Table 2. Masses of the CONSTOR® V/TC package and its components

Mass of the cask	Aapprox. 110 metric tons
Mass of an impact limiter	Aapprox. 20 metric tons
Mass of the puncture-resistant jacket	Aapprox. 31 metric tons
Mass of the transport package	approx. 181 metric tons

Drop Test Facility

The HAC free drop took place at the newly built BAM drop test facility at Horstwalde, near Berlin. The new test facility is constructed for test objects with a mass up to 200 metric tons. A 36m-high drop tower provides the capability for lifting and dropping in any desired orientation from a height of 9 m (30 feet) or more. Lifting is performed with an electric chain hoist at the top of the tower The maximum hook height is 30 m (98 feet). The impact target is constructed of a 2,450 metric tons reinforced concrete block (14 m / 46 feet x 14 m / 46 feet x 5 m /16 feet) faced with a steel plate (10 m / 33 feet x 4.5 m / 15 feet x 0.22 m / 8.7 in) of 77 metric tons as impact pad.

This foundation for drop tests is an "unyielding target" according to the advisory material for the IAEA Regulations. The basic requirements are that the stiffness and the mass are equal or higher than these of real foundations (for example soil, concrete, rock, etc.). An important criteria is that the foundation exhibits a mass that is at least ten times higher than the mass of the transport package specimen. A further advantage is the comparability of the results by performing each drop test on the same target. The drop test was carried out at a test cask temperature that corresponded to the ambient temperature on site.

Figure 8. BAM drop test facility in Horstwalde

Instrumentation of the Cask and Measurement Program

The cask was equipped with strain gauges and acceleration sensors to measure strains and accelerations at pre-selected cask locations during the tests. These measurement points were determined considering the maximum stresses to be expected according to the preliminary calculations.

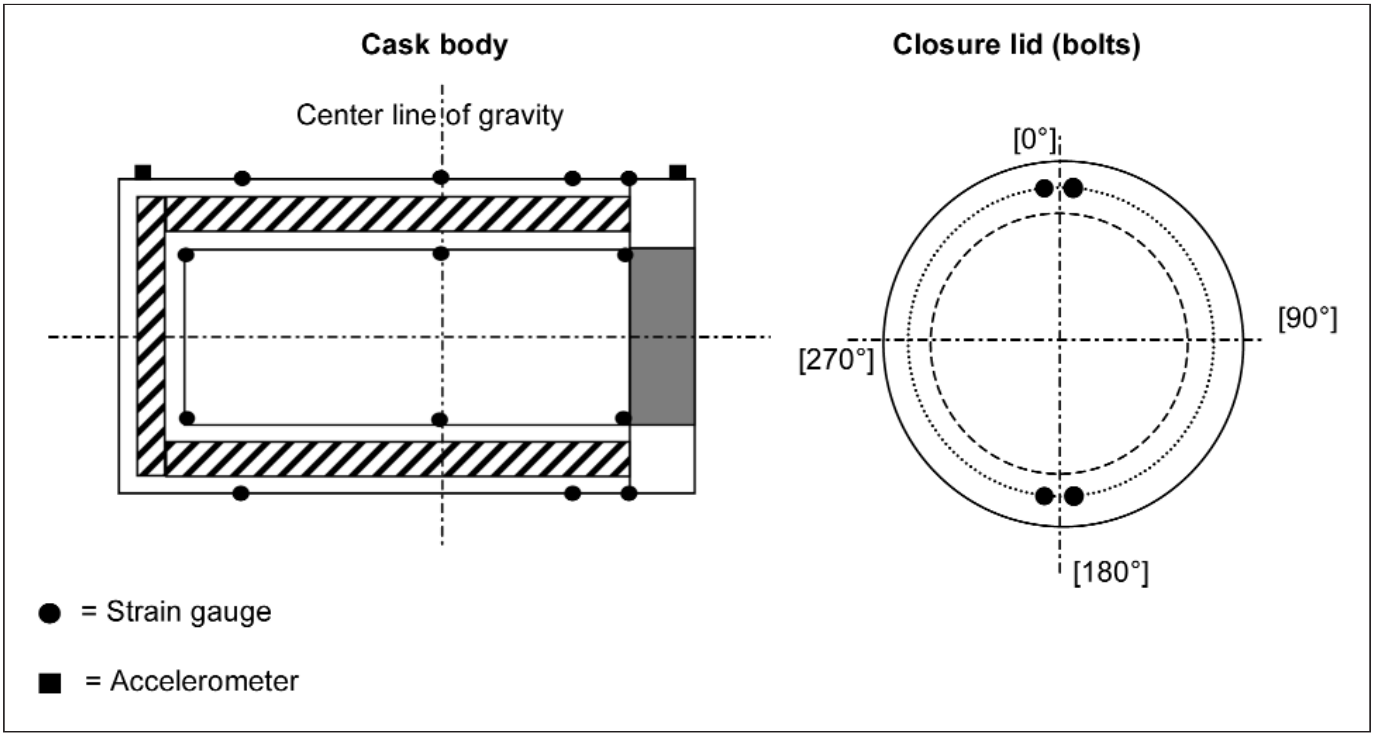
The inner liner, together with the head ring and the closure lid, forms the leak-tight containment of the cask. Therefore, primarily the weld seams between the inner liner and the bottom as well as those between the inner liner and the head ring are instrumented with strain gauges.

With regard to the lid system, the behavior of the closure lid bolts is of particular interest. Several bolts are fitted with three strain gauges at 90° intervals; these should provide information on the bolt strains during the drop tests onto the lid side. As the shielding lid has only a shielding function, no additional measurements are carried out on it. Acceleration measurements are used to verify the impact limiter design with regard to the effective drop energy absorption and therewith to the reduction of the deceleration on the cask and the inventory.

The recorded test data are subsequently processed (filtering of the time-dependent data and conversion of the strain values into stress values) and then compared with the results from the pre-test calculations. The strain measurements provide the essential component for the verification of the design methods.

The instrumentation plan for the HAC free drop test at Horstwalde is shown in Figure 9.

Figure 9. Instrumentation plan for the horizontal HAC free drop test at Horstwalde



Pre- and Post-Test Inspections

Before the drop test, all component dimensions of the transport package were measured (measurement record of the manufacturer) to be able to evaluate exactly the deformations after the test. There was also a leak-tightness test performed on the closure lid to make sure that the containment was in the proper condition before the test.

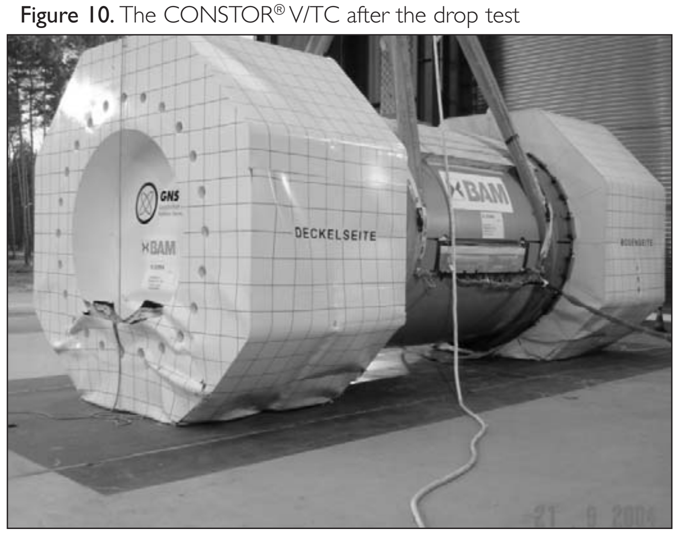
After the test, a visual inspection of the entire cask was carried out. Here, all deformations and any damage were comprehensively documented by photographs. Any deformation of the impact limiters the puncture-resistant jacket and the cask have been measured, as well as the leak-tightness of the closure lid.

Documentation and Evaluation of the Drop Test

A substantial amount of documentation had to be provided after the drop test. The post-test inspections were accompanied by quality assurance staff members of GNS and independent experts (TÜV) and a record was prepared for the drop test.

The strain and acceleration measurements are documented by BAM. These results are currently being used for the comparison with the outcomes of the pre-test calculations.

Figure 10. The CONSTOR® V/TC after the drop test

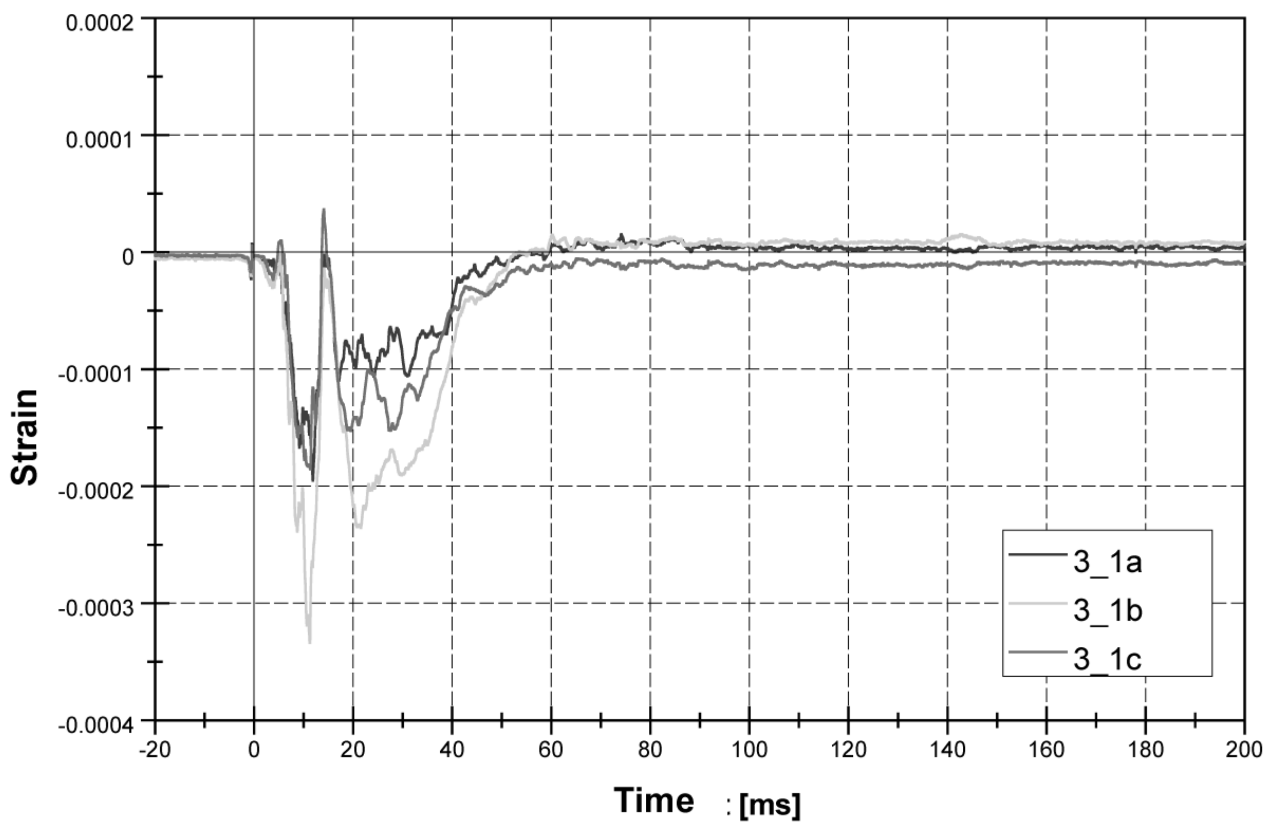


Test Results

Visual Inspection

Plastic deformation could not be observed or measured, either at the cask body or at the puncture-resistant jacket. The type and extent of the plastic deformation of the impact limiter plates and the wood corresponded with the expected damage according to the pre-test calculations. The deformation profile at the lid and bottom impact limiter were nearly identical (see. Figure 10).

Figure 11. Example of the recorded strain/time curve from a strain gauge at the inner liner



All bolted joints remained intact, those between the two parts of the puncture-resistant jacket as well as those between the impact limiters and the puncture-resistant jacket. Plastic deformation occurred only at the bolt threads between impact limiters and puncture-resistant jacket in the impact area.

Leak-Tightness Test

The leakage rate after the drop test did not increase. It corresponded to the result of the leak-tightness test that had been performed before the drop test. The requirement specification could thus be maintained.

The following values were achieved for the leakage rates at the two metal seals at the closure lid:

Seal 1: < $2{,}4 * 10^{-10}$ Pa m^3 s^{-1}
Seal 2: $8{,}7 * 10^{-9}$ Pa m^3 s^{-1}

Accelerations of the Cask Body and the Inventory Dummy

A detailed analysis of the results is currently being performed. As a preliminary result, a maximum value of about 75 g could be observed, which is well below the design value of 85 g.

Strain Values of the Cask Body and Lid Bolts

All strain gauges remained intact during the drop test and delivered strain values of the cask body and the lid bolts. Figure 11 shows an example of a recorded strain/time curve during the drop test. This strain gauge at the inner liner delivers a maximum stress of about 50 MPa. A detailed analysis comparing the values of the pre-test calculation with those values obtained by the measurements during the drop test will be completed by April 2005.

Summary and Conclusions

The CONSTOR® V/TC maintained its integrity and the closure lid remained leak-tight after the horizontal HAC free drop in Horstwalde. According to the current status of knowledge, the mechanical design of the CONSTOR® V/TC prototype will be well confirmed by the test results. Final conclusions can be drawn after the post-test calculations are completed.

With regard to further full-scale demonstration drop tests, two puncture tests in horizontal position are of special interest concerning the mechanical design of the package: one onto the center of the puncture-resistant jacket and one onto the lid or bottom side, beneath the area that is protected by the jacket.

Other drop orientations are already considered by the numerous tests with the half-scale models CONSTOR® VB-1 and CONSTOR® VB-2 that were referred to at the beginning of this paper.

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Acknowledgements

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Addressing Public Concerns about U.S. Spent Fuel Transportation: Lessons from Abroad

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Abstract

This paper discusses the international successes of spent fuel transportation worldwide, public reaction to such transportation, and proposes how such experience may be applied in dealing with perceptions of the safety of spent fuel transportation in the United States This discussion, while not comprehensive, addresses a number of specific issues that are raised in the United States by concerned individuals and entities, and examines international spent fuel transportation experience relative to those concerns. In addition, this discussion examines successful interactions with stakeholders about spent fuel transportation and the positive effects of such interactions.

Introduction

There has been significant discussion over the last several years about the challenges of the upcoming large-scale transportation campaign envisioned within the United States for shipping spent fuel to a geologic repository. Concerns about embarking on such a large-scale campaign to transport spent nuclear fuel across the United States have been expressed by some members of the public. Industry experts often cite that the safety of spent nuclear fuel transport is demonstrated by the large body of international spent fuel transportation experience, noting that significant amounts of spent nuclear fuel have been safely transported worldwide for almost forty years without a single accident resulting in a release of radiation. However, concerns have been raised as to whether this international experience is applicable to the U.S. transportation program due to the different modes of transport, length of rail shipments, and other variables. This paper examines, in light of relevant international experience, a number of specific concerns expressed by the public. This discussion demonstrates that the anticipated U.S. spent fuel transportation program is safe and that international experience is highly applicable to the U.S. program. This international experience also provides a suggested approach to gaining stakeholder acceptance in the United States.

Specific Issues

The following sections discuss specific issues raised relative to the safety of a large-scale spent fuel transportation campaign in the United States, as illuminated by comparable international experience. No full-scale cask testing U.S. Nuclear Regulatory Commission (NRC) regulations governing the licensing of transportation packages, provided in 10 CFR Part 711, designate tests for which the effects on a package are to be evaluated. The regulations allow for testing by subjecting a specimen (i.e., full-scale package) or a scale model to the specific tests, or another method of demonstration acceptable to the NRC. One such method is commonly referred to as "design by analysis," which typically consists of analytical evaluations of the prescribed tests, supported in many cases by some limited testing. Such limited testing often includes the energy-absorbing devices (a.k.a. impact limiters) and neutron absorbers or shielding materials. Some scale-model testing, usually on the order of 1/4-scale, is used for demonstrating performance of energy absorbers and cask response for certain dynamic testing conditions, including drop tests.

The current generation of spent fuel transportation casks licensed in the United States have been approved based on mostly analytical means. These packages have not been subjected to full-scale cask testing. The use of "design by analysis" is based on a history that did in fact include cask testing as a means of demonstrating cask performance. As more sophisticated analytical tools have become available, these tools were used in conjunction with the historical cask-testing results to provide a reasonable, scientifically sound basis for approval of these transportation packages. Full-scale cask testing has been performed internationally, most recently at the German competent authority BAM's full-scale cask drop testing facility at the Horstewalde Research Facility. The first of these tests was performed on a full-scale package of the CONSTOR® design, which was the largest cask ever subjected to a full-scale drop test, weighing in at an impressive 180 metric tons. Initial verbal reports from GNSI officials indicate that the preliminary results show that the analytical predictions are in agreement with the actual test results.

Other full-scale cask tests have been performed in the past decade, including tests of steel and ductile cast-iron casks. In each case, the analytical predictions of the cask behavior adequately predicted the actual test results. The accuracy of "design by analysis" as confirmed by these full-scale cask tests suggests that future full-scale testing is unnecessary. Full-scale testing would serve as an expensive demonstration of what the nuclear industry already knows to be true: the design of these casks is adequate to withstand the regulation-mandated design requirements.

Full-scale cask testing for purposes of public demonstration may be of some value, however, as part of an overall campaign to educate the public about the excellent safety of spent fuel transportation. Regulatory test conditions do not bound real-world accidents. NRC regulations define the conditions that a spent fuel transportation cask must be able to withstand, as prescribed in 10 CFR Part 71. The more extreme of these tests include a nine-meter (30 foot) drop onto a flat, essentially unyielding horizontal surface in any orientation, and a fully engulfing fire with an average temperature of at least 800°C (1,475°F) for thirty minutes. It has been suggested that these regulatory conditions are not adequate to cover real-world conditions, and that additional tests for more realistic conditions should be performed.

It is likely that the call for more realistic tests is due to a misunderstanding of the test conditions specified in 10 CFR Part 71. Relative to the 9-meter drop test, it has been pointed out that there are many bridges and overpasses that are higher than 9 meters. Also, the velocity at impact from nine meters is about 44 feet per second, or 30 mph, which is significantly slower that a conveyance (train or truck) may be moving. Thus, a real-world accident would involve heights and impact speeds that exceed those in the egulations. The regulatory drop tests are designed to provide a level of assurance that the casks are safe and can withstand any credible real-world accident. As such, they include requirements that are beyond those encountered in the real world. For the drop tests, this includes the requirement that the drops be performed onto an essentially unyielding surface. This means that the surface is hardened so that essentially no energy from the impact is mitigated by movement (flex, crush, or bending) of the impact surface; it all must be accommodated by the transportation cask package. This is extremely conservative compared to any real-world accident scenario, as even asphalt and concrete paving are much more flexible than the regulatory test surface and absorb a significant amount of energy.

To assess the amount of energy absorbed by various impact surfaces, studies were performed by Electric Power Research Institute[2] (EPRI) and Lawrence Livermore National Laboratory[3] (LLNL) in the 1990s. While the studies were performed with the purpose of identifying the energy absorbed by the deformation of the impact surface due to a drop of a bare cask (i.e., a cask without impact limiters) for various surfaces, the results show that even well-paved surfaces absorb a significant amount of energy.

Another "real-world" test involved the impact of a locomotive into a transportation cask at about 75 mph. The test resulted in minor damage to the cask, but no loss of cask integrity. The locomotive didn't fare as well; it was essentially demolished. The German authorities undertook a significant "beyond regulatory" test that involved dropping a cask from a height of more than 800 meters. Despite hitting the ground at a velocity in excess of 400 km/hour, the cask integrity was maintained. Other tests included a railcar gas tanker explosion and an aircraft impact simulation. In all of these tests, the integrity of the cask confinement was maintained,[4] such that a release of radiation would not have occurred.

Regarding the regulatory fire test, the Baltimore Howard Street Tunnel fire in 2001 is often cited as an example of a condition that is not bounded, since the fire was in a confined space and lasted a long time. Significant work has been performed by the NRC and others to estimate the conditions in the Howard Street Tunnel and temperatures reached in the fire through forensic investigations. The results of a 2003 NRC evaluation[5] concluded that, in fact, the temperatures that would have been achieved in a rail cask exposed to the estimated conditions of the Howard Street Tunnel fire would have resulted in the spent fuel cladding temperatures remaining below those that would cause breach of the cladding, and that containment would have been maintained, thereby maintaining confinement of all radioactive materials as required by the regulations. Damage to the cask shielding was also evaluated, and it was determined that this damage would have been less than or equal to that assumed as part of the regulatory fire analysis.

As shown in these real-world situations, the conditions prescribed by NRC regulations, while appearing not to bound potential real-world conditions, do in fact provide a robust assurance of the ability of these casks to withstand credible real-world events.

The large number of shipments will increase the likelihood of an accident. As is the case for all transportation, accidents happen every day on our highways and railways. There is no question about it — there will be accidents involving spent fuel transportation casks. What is important is that such an accident does not mean a release of radiation. There have already been accidents involving spent fuel transportation casks, but none of these accidents have involved a release of any radiation. The cask designs are robust and are able to survive real-world accident events and maintain their integrity.

There are those who think that, as the number of shipments increases, inevitably an accident will result in a release of radioactive material, that "it is just a matter of time." This conclusion is based on an assumption of inevitability of an event or sequence of events that will exceed the capability of the cask and result in a release of radiation. While it is impossible to say such an event would never occur, the history of transporting spent fuel worldwide has demonstrated an excellent safety record.

International experience, including truck, rail, and ship transport over the last thirty-eight years, with more than 70,000 MTU shipped, has never resulted in an accident that resulted in a release of radiation. BNFL has shipped more than 35,000 MTU within the UK and has transported spent fuel more than 16 million total miles worldwide. The applicability of international spent fuel transport experience to anticipated U.S. transportation operations has been questioned due to the differences in the principal modes of transport. International transport includes many miles at sea, while U.S. transport will be performed mainly by rail. It has also been pointed out that the rail transports overseas are of shorter length than those from most U.S. nuclear sites to the anticipated repository site.

It is noteworthy, however, that European rail transport occurs almost exclusively in relatively highly populated areas, while much of the anticipated rail mileage for U.S. transport will be in much less populated areas. This difference suggests that U.S. transport is less risky than international transport, in terms of the number of people who could potentially be affected by a given accident. Furthermore, Direct Rail Services (DRS) in the UK reports annual total shipments of 264 million gross MT-miles, all without incident. This is equivalent to about 90–100 shipments per year from the East Coast of the United States to the proposed repository site. These DRS shipments include transport of spent fuel through London an average of five days per week. Clearly, this international experience is relevant to, and in some cases even bounding of, U.S. transportation plans. In the UK, rail shipments of spent fuel and other radioactive materials are made to Sellafield on an almost daily basis, passing the nearby town of Drigg. In a recent visit to the UK, discussions held by U.S. scientists with local residents indicated a very low level of public concern with the shipments. These rail shipments have been occurring for so long, without a single incident of radiological release, that the public has little cause to think about the shipments. A local resident, when asked if he is concerned about shipments of spent fuel passing through his town, responded, "No. Why, should I be?" This attitude toward nuclear fuel shipments is akin to the attitude of many Americans when it comes to everyday dangers that they take for granted. Most people don't think twice about the propane tanker on the freeway next to their car. Railcar accidents resulting in chlorine gas leaks do not result in a call to end the use of chlorine. Why? Because we are used to these risks and accept them. Spent fuel transportation is something that UK residents near Sellafield are accustomed to and accept. Many of the public's concerns can be assuaged by adequate factual information, but there is no substitute for experience.

Local responders to an accident won't be able to deal with it. The concern has been expressed that standard emergency response personnel will not be able to effectively respond to an accident involving a spent nuclear fuel cask because they will be unable to assess the radiological conditions associated with the accident. As a result, many assume that first responders will require special training, equipment, and other items in order to conduct spent fuel transport on a large scale in the United States. In the UK, this is not an issue. All local responders have basic training in placards (as do responders in the United States) and protocols are in place for how to deal with an accident involving a radiation-placarded vehicle. Typically this involves setting up a perimeter and contacting the local knowledgeable authority. In the UK, this is often the nearest nuclear plant.

This approach is not unlike the approach used in the United States for accidents that are suspected of involving hazardous substances. On any given day, a highway is often closed due to an accidental spill of an "unidentified white powder," and actions are taken to set a perimeter and contact the appropriate responder. As demonstrated internationally, there is no need for an elaborate response system for first responders. There are specific responders trained to deal with radioactive materials shipment accidents, similar to those who respond to accidents involving hazardous materials.

Specifics about terrorism associated with spent fuel transportation casks are considered to be safeguards information and as such are not dealt with in this paper. However, it can be noted that the NRC, U.S. Department of Homeland Security, and other entities are investigating possible terrorist activities and the ability of these casks to withstand such activities. A determination of an unacceptable vulnerability would likely result in new or revised requirements or other mitigating actions.

It is worth noting that spent fuel transport casks are unattractive targets for the purposes of causing terror. The casks are very robust and hard to breach, and, in the event of a breach, it is difficult to actually disperse the radioactive material. There are many other targets that offer more dramatic, terror-inducing results, and it is extremely unlikely that a terrorist would target a spent fuel transportation cask.

Applicability of International Experience of Stakeholder Interactions in the United States

BNFL's extensive experience in dealing with stakeholders, including utilities, government agencies, non-governmental organizations (NGOs), and local officials, has resulted in highly successful transport of spent nuclear fuel on a routine basis. Open communication with the public and the sharing of relevant and factual information is the key. BNFL uses a well-publicized public outreach program, which includes an open process for addressing stakeholder concerns.

These include literature, public meetings, and in some cases development of joint task forces that include the concerned stakeholder parties.

This approach provides a means for the stakeholders to be actively involved in the process for addressing concerns, and to be involved in the designation of studies or other activities done to investigate questions, and therefore ownership of the resulting conclusions and actions taken. This inclusive approach results in public acceptance of the spent fuel transportation program, even though some of the concerned stakeholders may continue to maintain their opposition to nuclear power. This suggests that the best approach for stakeholder interactions is not to try to change someone's mind about nuclear power, but rather to get the individual involved in the process. Through this involvement, people learn that the approaches and solutions to the ransportation issues are safe and effective, thus assuaging their concerns.

Summary

Spent fuel transportation is safe. International experience, including truck, rail, and ship transport over the last thirty-eight years, includes more than 70,000 MTU shipped more than 16 million miles with zero accidents resulting in a release of radiation. While some may argue that ocean shipment of spent fuel is not the same as rail shipment, and that, since U.S. shipments to the repository will be mostly by rail, such international experience is not applicable, such arguments do not acknowledge the facts. Almost all spent fuel that has been transported by ship begins or ends its journey on a railcar or truck. This rail and truck transport is highly relevant to U.S. transportation plans.

BNFL's approach of stakeholder involvement in an open, interactive process has proven very successful to obtaining public acceptance for spent fuel transportation. Such an approach could be successful in the United States to obtain stakeholder support, or at least acceptance, of the spent fuel transportation program. As this program gets underway, ongoing, safe transportation activities will result in these transports eventually being accepted as routine.

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Qualification Test of Concrete Cask Performance in CRIEPI

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Note: This work was executed under contract with METI/NISA.

Abstract

In 2003 CRIEPI successfully concluded a four-year study program of demonstrative tests for the interim storage of spent fuel that is mainly related to concrete cask storage technology. The program was aimed at the creation of away-from-reactor dry storage by 2010. A concrete-cask storage system is considered to have an economic advantage. The acknowledgements obtained in these tests reflected Japanese safety guidelines for concrete casks issued by METI/NISA (Ministry of Economy and Trade Industry/Nuclear and Industrial Safety Agency) in June 2004. This paper introduces the qualification test of concrete cask performance of spent fuel storage technology executed in CRIEPI.

Introduction

According to the policy of Japan, spent fuel generated by nuclear power plants (NPP) is designated as useful recycled resources and shall be properly stored until reprocessing. Soon, the quantity of spent fuel stored at each NPP site is going to increase due to the ending of an oversea reprocessing contract and the delay of a domestic reprocessing project. Therefore, the construction of interim spent-fuel storage facilities at NPP sites or away from NPP sites is expected to begin soon. The dual-purpose metal cask that can be used for storage and transportation before and after storage has received the highest priority in implementing storage facilities in the short- and medium-term because of its superb economics compared to water pools. In Japan, dry-storage facilities using metal casks have been operating at the Fukushima-Dai-Ichi site of the Tokyo Electric Power Company and the Tokai-Dai-Ni site of the Japan Atomic Power Company.

For the long-term, research is ongoing for the concrete modular dry storage technologies, with the aim of developing better economic performance. Key issues in this research include safety standards in operation and maintenance during storage and unloading/loading for transportation, the long-term integrity of metal canister and concrete materials, and so on. In 1997, CRIEPI began a new research program—the demonstration test for interim storage of spent fuel mainly involving concrete cask storage technologies, with the aim of obtaining basic data for regulating safety.

This paper summarizes the qualification tests of concrete cask performance of spent fuel storage technology.[1,2]

Demonstration Program for the Qualification of Concrete Cask Performance

In the demonstration program, the following studies (see Figure 1) were completed.

a) For concrete material and structures
   i) The long-term durability of concrete material (carbonation and salt damage)
   ii) The dynamic strength of concrete materials under high temperatures and in the event of an accident
   iii) Characteristics of heat transfer and cracking due to thermal stress
   iv) Shielding performance of concrete structures
b) For metal canisters
   i) Impact and corrosion resistance of multi-purpose canisters with welded components
c) For spent fuel
   i) Development of a nondestructive monitoring method
   ii) Characteristics and long-term performance of high burn-up and MOX spent fuel
d) A demonstration program for determining concrete cask performance (A schedule of this demonstration program is shown in Figure 2.)
   i) Basic design of Japanese concrete cask. Two types of concrete casks, a reinforced-concrete cask (RC cask) and concrete-filled-steel cask (CFS cask) to store the high burn-up spent fuel, were designed.
   ii) Manufacture. Two types of full-scale concrete casks and multi-purpose canisters were manufactured.
   iii) Demonstration tests. Heat removal tests of the concrete cask were executed taking into consideration normal, off-normal, and accidental events, and well as impact tests on the metal canister. Seismic tests using a scale-model cask and streaming tests with the air duct components were carried out.
   iv) Safety analysis. Safety analysis was performed using the information obtained in the demonstration tests to contribute to safety standards for concrete modular structures, systems, components.

Figure 1. Schematic showing performance of reinforced concrete components in dry storage

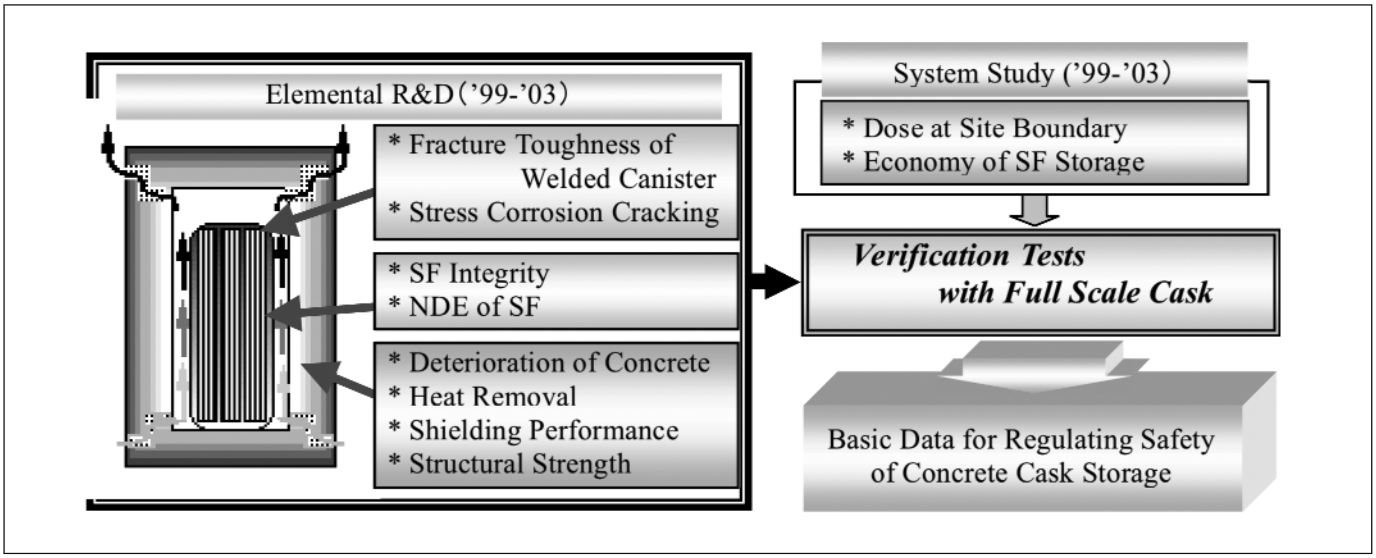
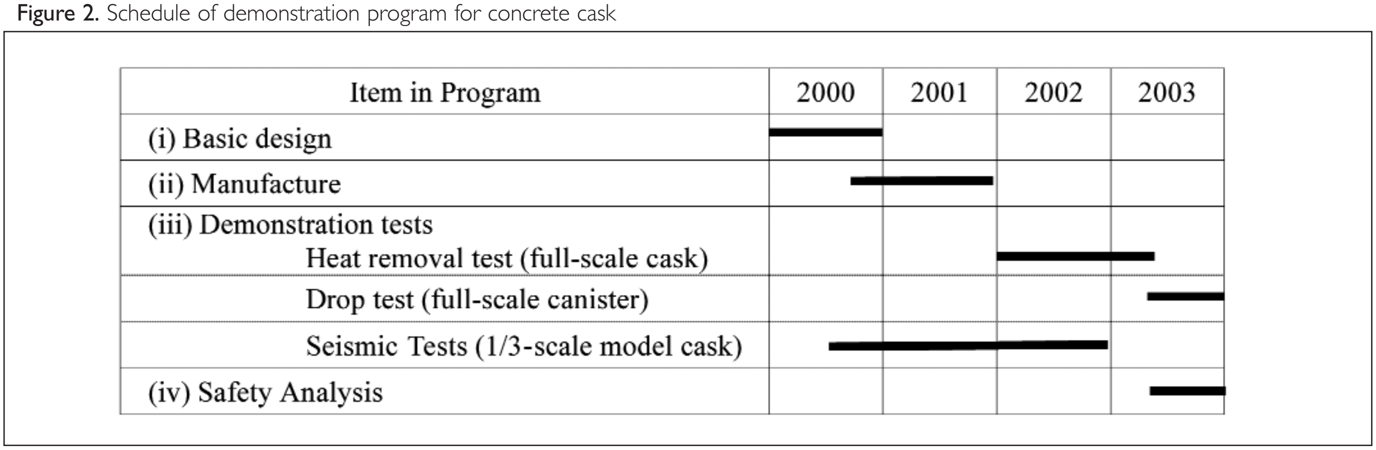


Figure 2. Schedule of demonstration program for concrete cask



Program of Demonstration Tests for Determining Concrete Cask Performance

Demonstration Test Facility in Akagi Test Center

The demonstration test facility as shown in Figure 3 was constructed in the Akagi Test Center of CRIEPI, about 130 km north of the center of Tokyo. In this facility, there is a heat removal test area and a drop test area.

In the heat removal test area, there are two movable tents on the rail. One tent is used for avoiding wind and rain for the preparation of the test and the other is used for the heat removal test. The heat removal test tent has an outer wall and insulated inner wall to decrease the fluctuation of ambient temperature. During the heat removal test, the concrete cask is located in the middle, and the cooling air goes inside through the four windows with louvers and the crevice between the base concrete and the wall, and goes outside through the ventilators attached on the roof as shown in Figure 3. The horizontal and vertical distances between the cask surface and the inner wall are about 2 m and 4 m, respectively. In the drop test area, a steel plate is fixed on the base concrete. The steel plate is 7.5 m long, 4.5 m wide, and 50 mm thick. The thickness and weight of the base concrete is 2 m and 400 tons, respectively.

Basic Design of Japanese-Type Concrete Cask

Strength and safety to the load must be maintained when considering the conditions under which casks are used (e.g., size of the site, installation on the shoreline, and seismic factors). In our country we are particular about the structure of the cask and the material used for the cask which we assume during the design phase. Preliminary design items and parameters are shown in Table 1. The concrete cask was assumed to be for indoor use.

Figure 3. Overview of the demonstration test facility in Akagi Test Center of CRIEPI

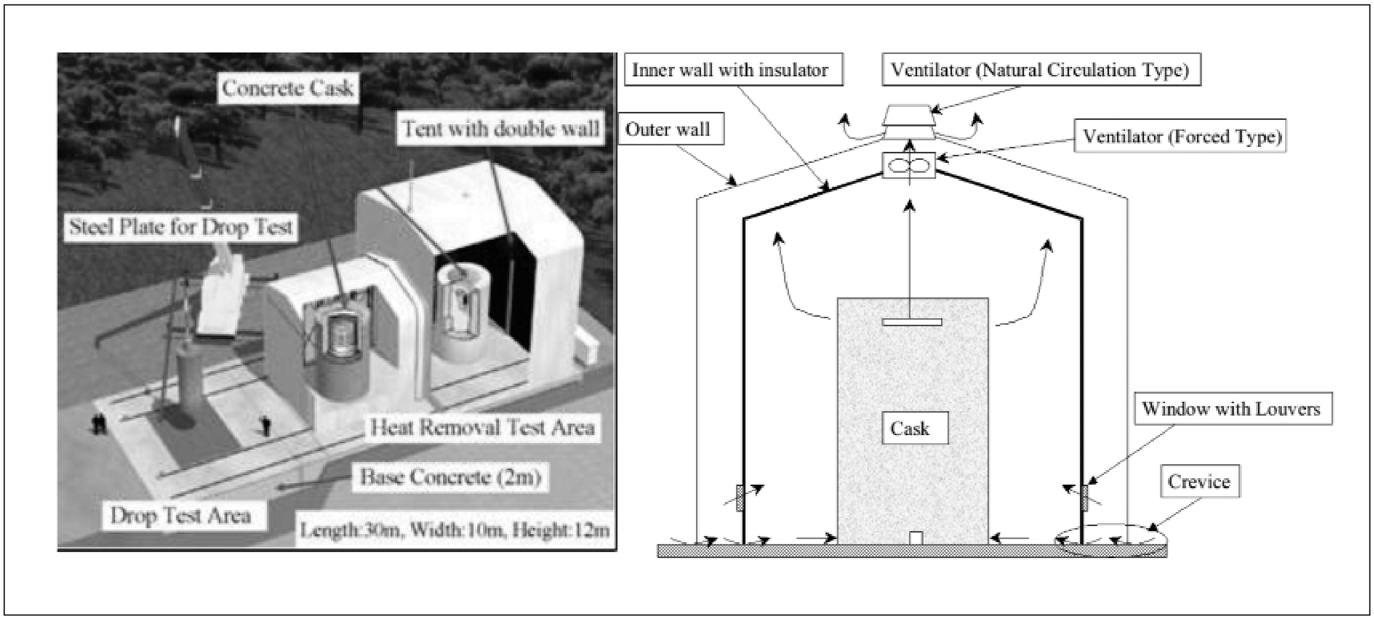


Figure 4. Outline of the concrete cask

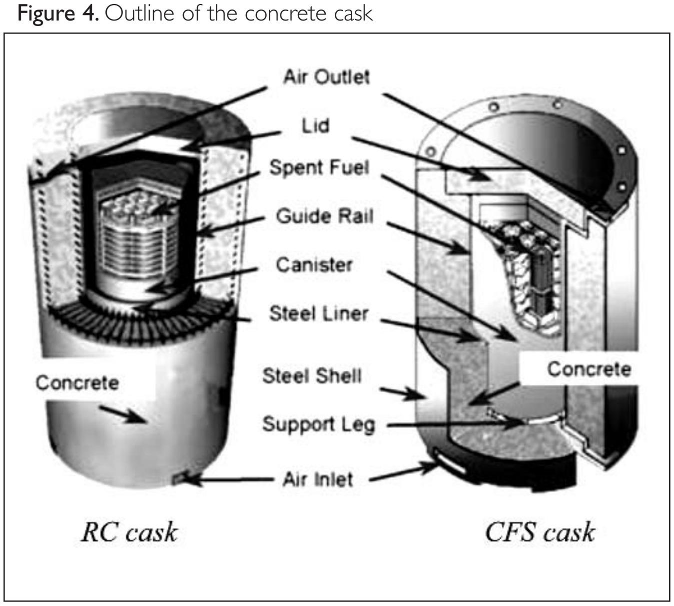
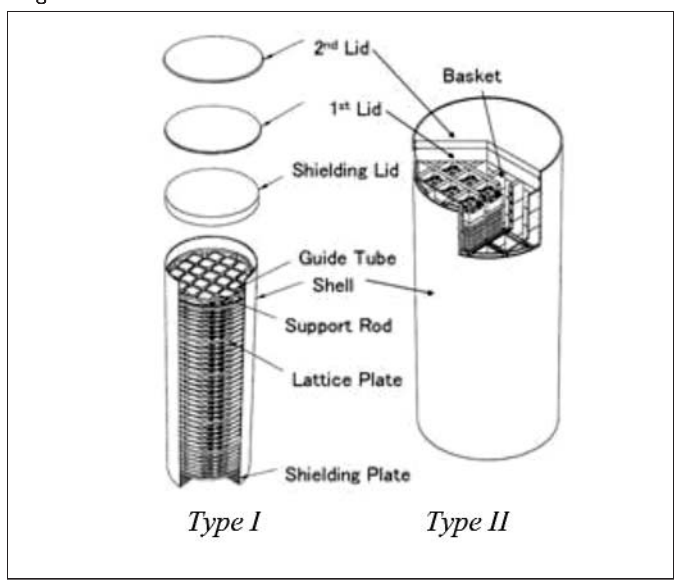


Figure 5. Outline of the canister



Preliminary designs for two types of casks, RC cask and CFS casks were employed as the basic structure as shown in Figure 4. The RC cask is made from a reinforced concrete storage container and the reinforced concrete becomes a structure strength part to the assumed load. On the other hand, in the CFS cask, the concrete storage container consists of concrete covered with a steel sheet, creating a steel structure; concrete is not the structure strength, it is a radiation-shielding material.

Two types of canisters were designed as shown in Figure 5. Each canister can store twenty-one PWR spent fuels, and for each canister body high corrosion-resistant material is used. The basket of type I consists of guide tubes and stainless steel plates. The stainless steel plate is fixed at constant intervals by steels rod and it has twenty-one square holes for the guide tubes. The guide tubes are placed in the holes and fixed to the plate. To increase thermal conduction, an aluminum plate is fixed to the stainless steel plate. The basket of type II is the assembly of a rectangular hollow block made of an aluminum alloy.

Table 1. Preliminary design item and parameter

Design Item	Condition	Evaluation Item
Thermal	Normal	Heat generation rate, air flow rate
	Off-Normal	Integrity of the fuel cladding, temperature
Containment	Normal	Quality assurance of the welded structure of the cansister
Shielding	Normal	Dose rates, cooling air activation
Criticality	Normal	Wet condition, dry condition
Structual Strength	Normal	Durability, thermal stress, Internal pressure, Seismic ability
	Off-Normal	Drop of canister, tumble of cask, Blockage of air inlet
Design sorage period	40–60 years	
Design parameter	Fuel type	17 x 17 array for PWR
	Enrichment (wt percent U^{235})	4.9
	Burn-up (MWd/kgHM) (max)	55
	Cooling time (year)	10
	Environmental temperature	33

Table 2. Specifications of the concrete cask

	Type of Storage Container	RC	CFS
Storage Container	Height	5,787 mm	6,120 mm
	Outside diameter	3,940 mm	3,800 mm
	Inside diameter	1,850 mm	1,838 mm
	Weight (without canister)	150 t	154 t
Canister	Type of Canister	Type 1	Type 2
	Height	4,630 mm	4,470 mm
	Outside diameter	1,676 mm	1,640 mm
	Weight (with spent fuels)	35 t	30 t
	Body	Steel	Stainless steel
	Basket	Stainless steel	Aluminum alloy

Fabrication of Full-Scale Concrete Cask

Based on the design, two types of full-scale concrete containers and canisters were fabricated for the demonstration tests. The main specifications of these casks are shown in Table 2. The ratio of reinforcement for the RC cask was 1.7 percent from the point of view of ensuring good durability for the long-term storage. We used high-quality concrete (water/cement ratio is less than 50 percent) including a highly efficient AE water reducing agent for the casks. Concrete containers were fabricated without the placing joint. For the CFS storage container, the studs are welded on the inner surface of the outer shell of the cask and we used the same high-quality concrete as described above. Figure 6 shows two types of full-scale cask and canister baskets for Type 1.

Heat Removal Test Using the Full-Scale Casks

The heat removal tests using two types of concrete casks were completed. In this section, the examples of the test results using RC cask are introduced.

Test Cask

During actual storage, two lids are welded to the canister body to maintain the confinement. However, during the thermal tests, only one lid is welded to the body, taking into account the opening of the lid after the test.

In the canister lids, there are twenty-one holes for heaters and three holes for measurements. The heater was inserted to a dummy weight steel structure and fixed on the top of the secondary lid by a flange that consists of a sheath heater and generates heat in the same length as the spent fuel.

Test Condition

Table 3 shows the test condition. Test parameters are heater power, closure rate of air inlet, and cask position. The canister is sealed and filled with helium gas at 0.1MPa approximately at the ambient temperature. In the beginning, the tests were performed in the vertical position and then the cask position was changed to horizontal. During the tests, the tent ventilator was in operation so that the stratification boundary does not go down to the level at the air outlet.

Measurement

The cooling air removes most of the heat discharged from the spent fuel in the concrete. As it is important to evaluate the heat balance, the air velocity, and the flow rate at the air inlet were measured precisely. While measuring the air inlet flow with the

Figure 6. RC cask, CFS cask and canister basket for Type 1

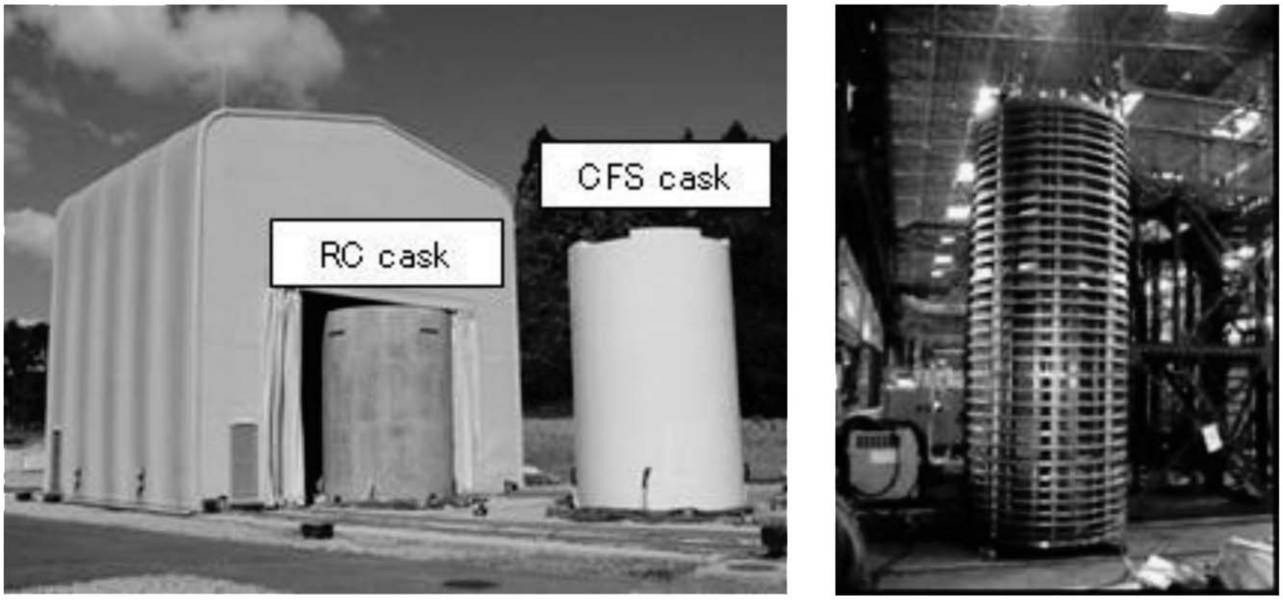


Table 3. Test Condition

No.	Cask Position	Cavity Gas	Total Heat Poser (kW)	Closure Rate of the Air Inlet (percent)	Situation
1	Vertical	He	22.6	0	Steady state
8	Vertical	He	16.0	0	Steady state
2	Vertical	He	10.0	0	Steady state
3	Vertical	He	22.6	50	Steady state
4	Vertical	He	22.6	100	Transient
5	Vertical	Leak condition	22.6	0	Transient
6	Horizontal	He	22.6	0	Steady state
7	Horizontal	He	22.6	100*	Transient

* In this case, all the inlets and outlets are closed.

anemometer, a rectangular pipe with bell-mouth structure was used to regulate the three-dimensional air inlet flow to a one-dimensional flow. As the temperature exhausting from the outlet duct is very high, the air velocity at the outlet duct is measured with a propeller flow sensor.

Test results

Case 1

(a) Flow velocity and temperature

In Case 1, the normal storage condition corresponding to the initial state during the storage is considered. The inlet air temperature was 23°C in the steady state. Average velocity is 0.84 m/s at the 180° inlet duct. The total flow rate of the cooling air is 0.28m3/sec and temperature increase of the bulk air is about 65°C.

(b) Concrete temperature

Figure 7 shows the temperature distribution of the concrete container. The temperature distribution along the radial direction is almost linear and the maximum concrete temperature around the outlet duct is about 81°C. As this value seems to exceed the estimated value obtained in the pre-thermal evaluation, it is necessary to modify the evaluation method and preliminary design.

(c) Canister temperature

Figure 8 shows the circumferential surface temperature of the canister compared to the pre-calculation value. The circumferential surface temperature in the 45° direction is lower than that in

Figure 7. Temperature distribution of the concrete container

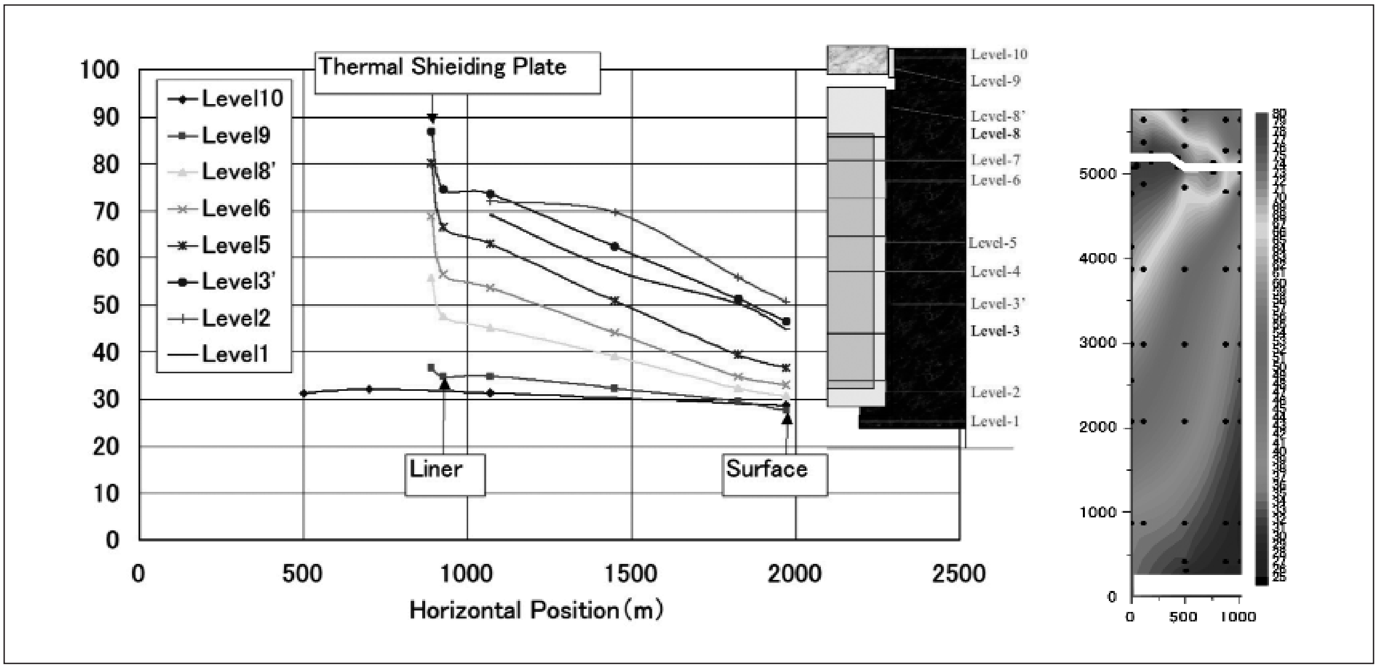
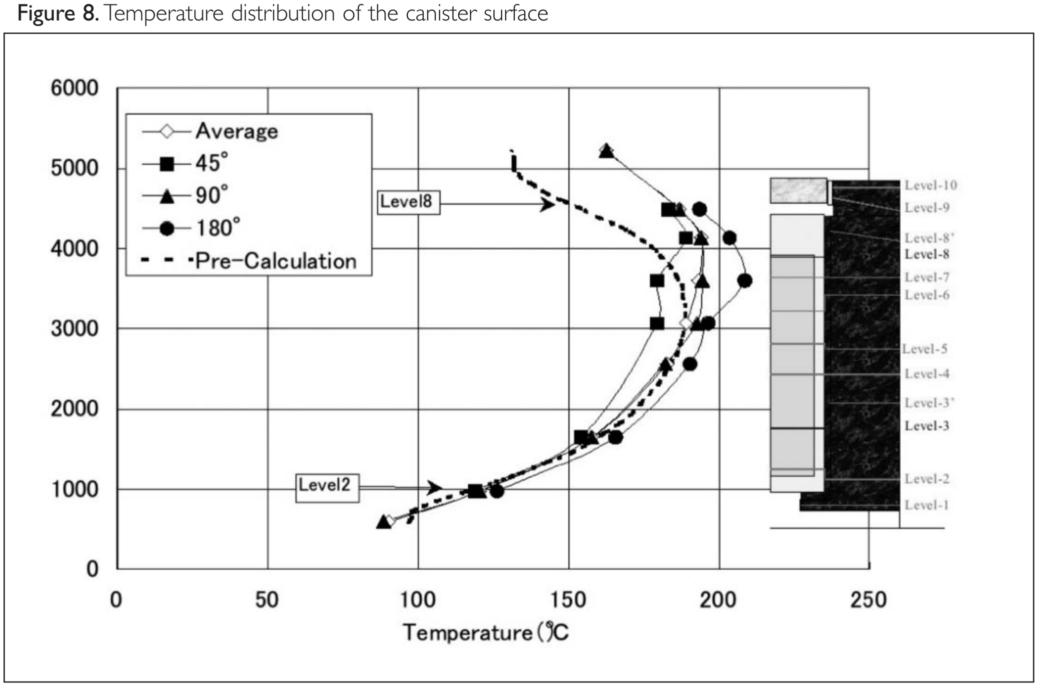


Figure 8. Temperature distribution of the canister surface



the other direction because of the contact between the canister and the guide rail, and furthermore, the basket may also have contact with the canister body in the 180° direction. Concerning to the longitudinal distribution of the surface temperature, there is not good agreement between the experimental value and the precalculation results, especially in the upper part of the canister. Because of this temperature difference, the temperature of the concrete container (around the outlet duct and the bottom of the lid) obtained in the pre-calculation value is considerably smaller than the test data. Therefore, it is very important to take account of the contact condition and the longitudinal heat conduction model in the preliminary design and evaluation.

Figure 9. Schematic diagram of the heat balance

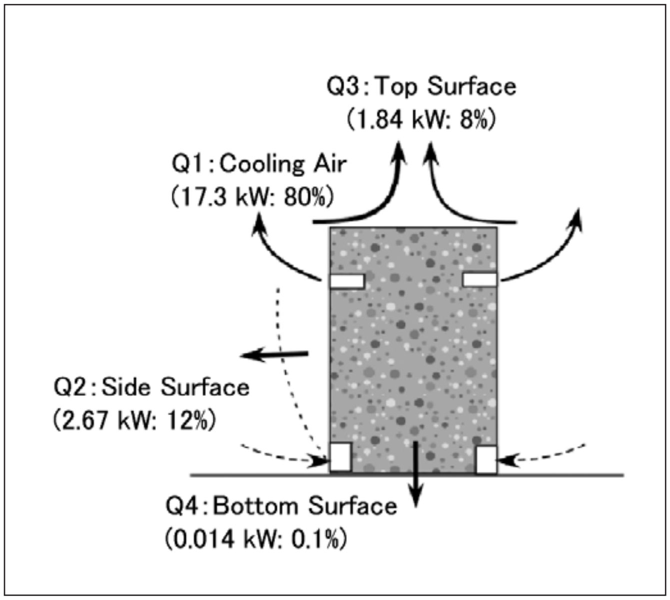


Figure 10. Crack on the top of the cask

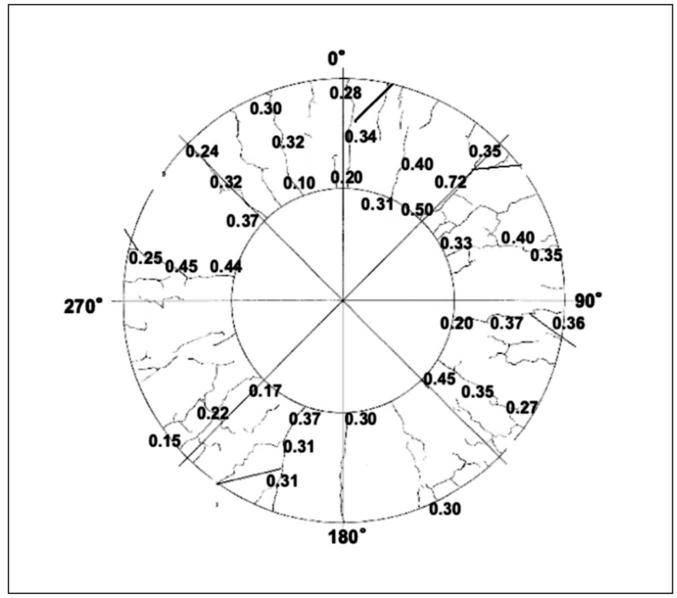


Figure 11. Relation between the temperature difference and the width of the crack

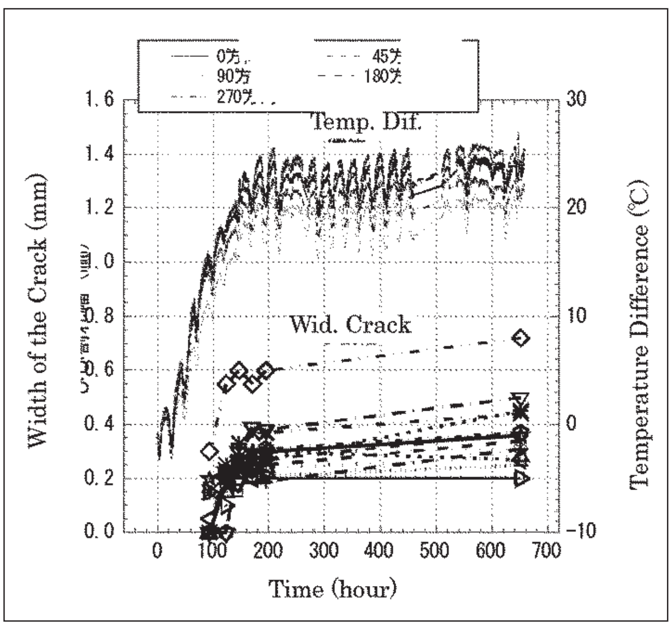
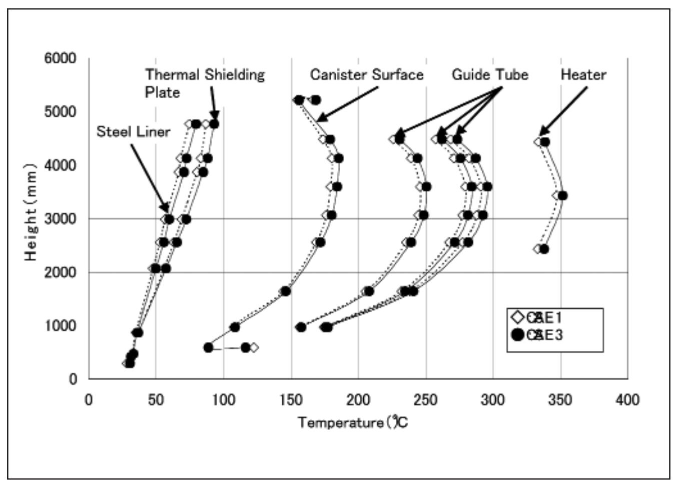


Figure 12. Comparison of temperature distribution in the axial direction between cases 1 and 3



(d) Heat balance

The heat discharged from the concrete cask to the environment is attained by the cooling air and heat transfer on the cask surface. In order to obtain the heat balance, the amount of heat removed by the air and heat transfer on the cask surface is calculated using air and temperature distribution data in the inlet and outlet ducts and temperature gradient in the concrete container. Figure 9 shows the ratio of the heat balance. Eighty percent of the heat is removed by the cooling air.

(e) Strain and crack

During the test, the concrete surface cracked. In the upper part of the cask, the number of cracks and their width are larger than that in the lower part. Figures 10 and 11 show the crack on the top surface of the cask and the relationship between the temperature difference and the crack width. The cracks occurs and the crack width increases as the temperature difference between inside and outside of the concrete container increases, and moreover tension stress appeared on the outside region, and compression stress appeared on the inside region.

Figure 13. Temperature distribution of the canister surface (comparison among cases 1, 2, and 8)

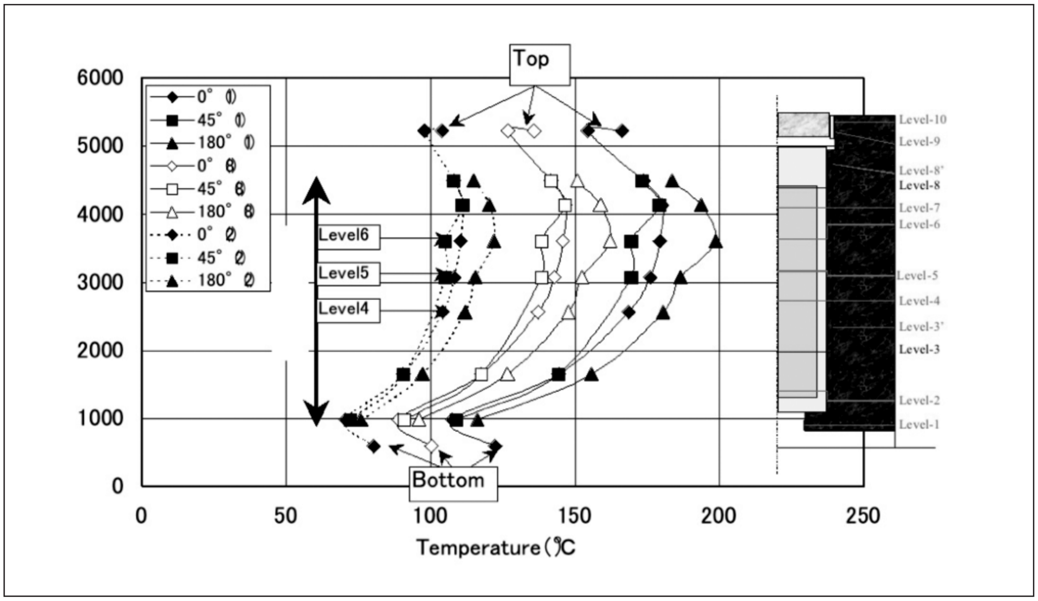
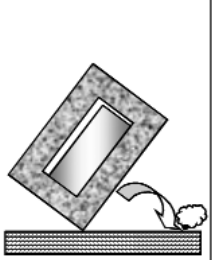
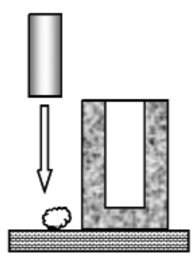


Figure 14. Drop test condition

Canister	Type I	Type II
Non-mechanical drop or impact events during handling	Tipping-over 	Drop 
Orientation	Horizontal	Vertical
Height	1m*	6m**
* Equivalent drop height for rotational velocity caused by tipping-over from height of GC ** Drop height from cask height		

Case 3

This case is the condition of 50 percent blockage of the inlet. After closing the inlet, the condition reaches the steady state. In this case, the air flow rate decreases and the air temperature of the outlet increases compared with the Case 1. Judging from the temperature distribution, drift flow in the flow area that effects upon the temperature of the cask has not been observed. As the temperature increase is only 5°C, the influence of the 50 percent blockage on the temperature seems to be small. Figure 12 shows the temperature distribution of the canister and the cask body in the axial direction by comparing the test results between cases 1 and 3.

Cases 2 and 8

For test cases 1, 2, and 8, heat power is considered as a test parameter. The data with small heat power is necessary to evaluate the condition in the middle and final state of the storage. Especially, as the temperature of the canister surface goes down by the heat power decrease, it is important to evaluate the cold part of the canister surface from the point of view of stress corrosion cracking.

Figure 13 shows the temperature distribution of the canister surface among test case 1, 2, and 8. According to these test results, it is found that temperature of lower and upper part of the canister is relatively low.

Drop Test

The demonstration drop test program using the double-lid welded multi-purpose canister (MPC) was executed, with the aim of obtaining basic data for regulating safety.[3] Figure 14 shows the drop test conditions. Two drop tests were conducted in horizontal and vertical orientations, considering non-mechanical drop or impact events during handling, and the drop height was 1m and 6m, respectively.

As for the object target, the hard target, namely, the 5 cm thickness steel plate attached to the concrete block (width 13 m, thickness 2 m, length 10 m, total weight about 550 tons) was applied. Moreover, to monitor the impact response of the target block during the tests, the accelerometers were set inside the concrete block.

Regarding contents of MPC, dummy steel structures equal to the total weights of the spent fuels (14.7 ton) were used.

Figure 15. Overall view of the horizontal drop test

Measurements
To estimate the impact forces and plastic deformations on the MPCs, acceleration and strain were measured at various points in the test MPCs during drop tests. Considering the impulsive vibrations from the impact of the test MPC on to the steel plate, a specified gauge-type accelerometer with large capacity (maximum 10,000 G) was applied. The measuring sampling rate was set to 1μsec and all of data were measured simultaneously without delay.

Horizontal Drop Test Results
Figure 15 shows photographs of the test canister before and after the drop test. The test canister was slightly deformed near the impacted area. Figure 16 shows time histories of accelerometers at various points in the test canister, measured in the drop test. The average deceleration value was about 436 G at the top of the lids.

Figure 17 shows the schematic view of He leak test for the second lid. He leak tests were performed before and after the drop tests to confirm the integrity of leak-tightness of the test MPCs (especially welded lids) against impact loads. Measured leakage rates shows the integrity of sealability at lids and canister shell, as all values are under 1.0×10^{-9} Pa*m^3/s.

In Figure 18, photographs of the cut section of the directly impacted welded part during horizontal drop test through microscope with magnified by 5.7 times are shown. Crack initiation could be found in this figure due to the impulsive moment around the top corner of the test canister. However, the initiated crack was arrested in the first welded layer.

Figure 16. Measured time histories of acceleration during horizontal drop test

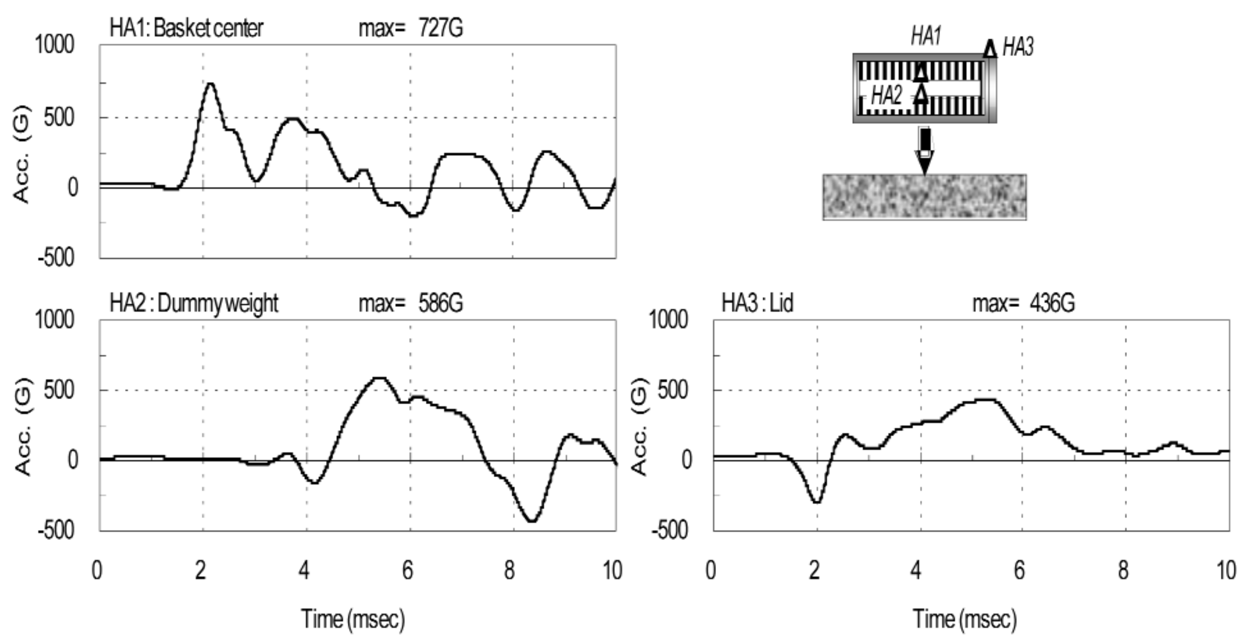


Figure 17. He leak test for second lid

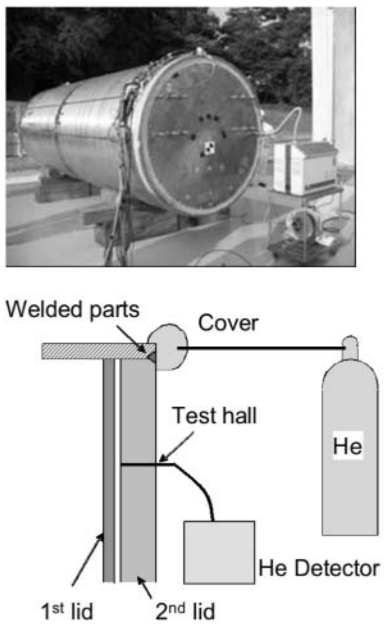


Figure 18. Magnified view of the cut section of the directly impacted welded part during horizontal drop test

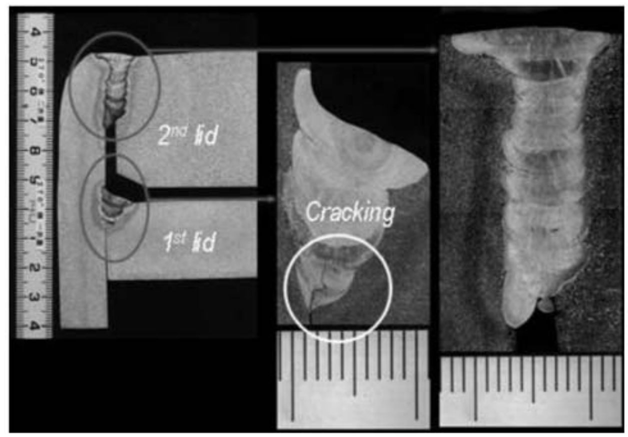
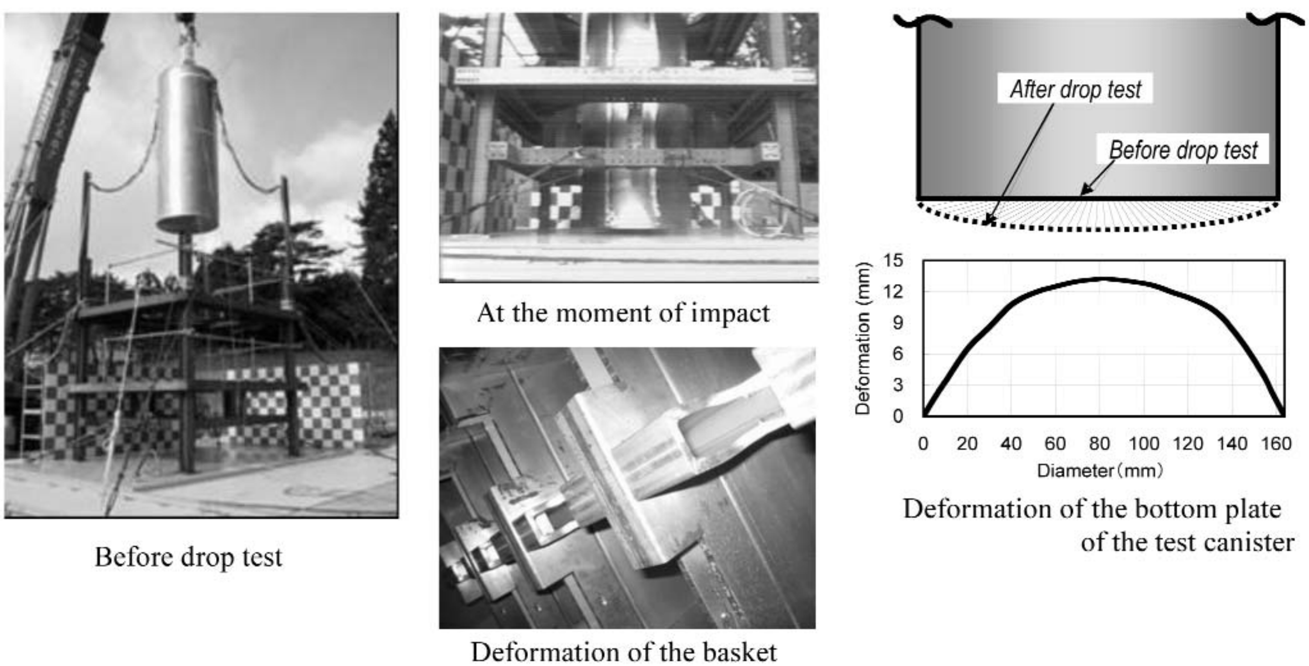


Figure 19. Overall view of the vertical drop test



Vertical drop test results

Figure 19 shows photographs of the test canister before and after the drop test. The bottom plate of the test canister was deformed by the force of inertia of the contents. However, the basket was slightly deformed near the impacted area. Figure 20 shows time histories of accelerometers at various points in the test canister, measured in the drop test. The average deceleration value was about 1,153 G at the center of the shell. He leak tests were performed before and after the drop tests to confirm the integrity of leak-tightness of the test MPCs (especially welded lids) against

Figure 20. Measured time histories of acceleration during the vertical drop test

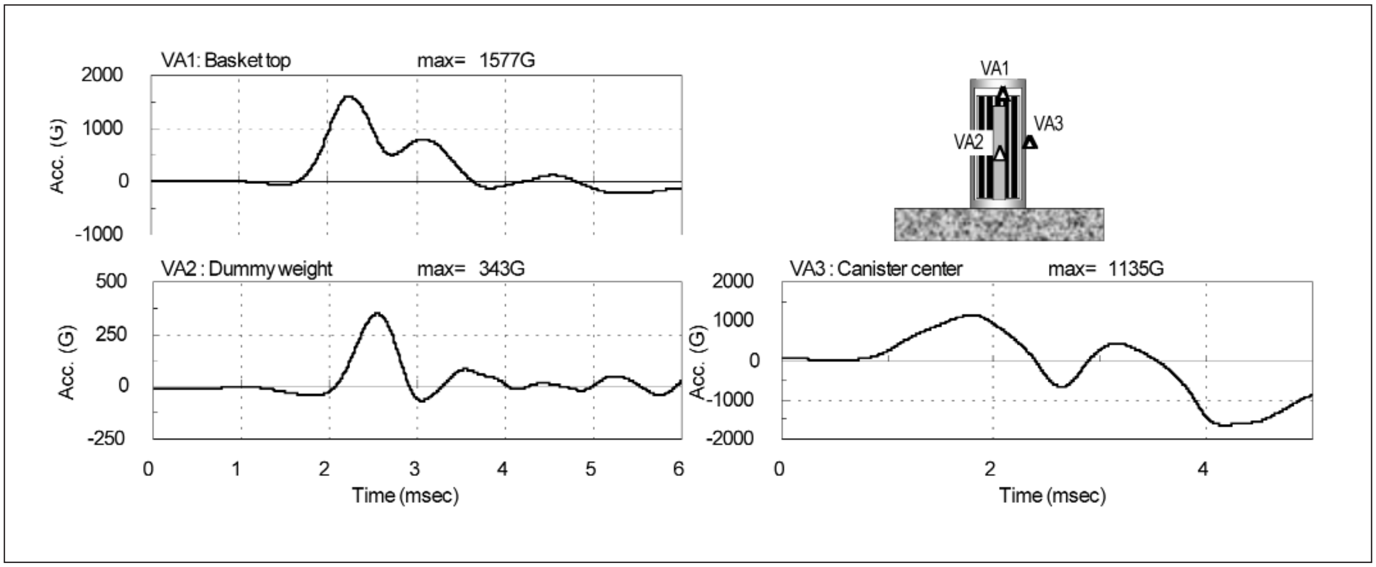
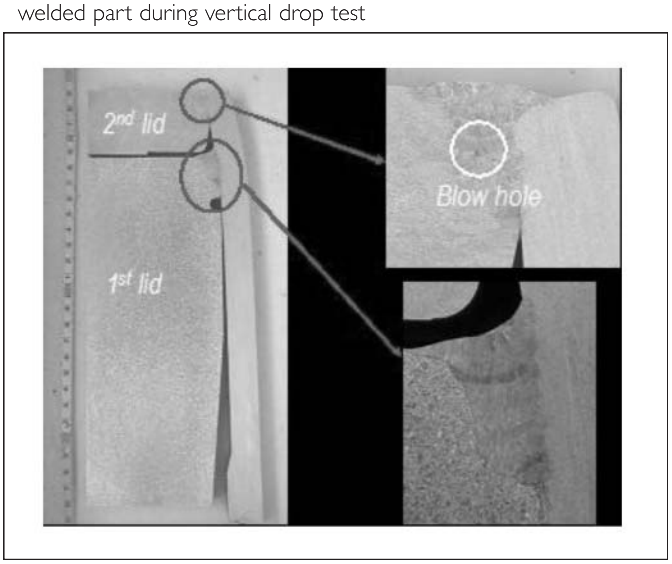


Figure 21. Magnified view of the cut section of the directly impacted welded part during vertical drop test



impact loads. Measured leakage rates shows the integrity of sealability at lids and canister shell, as all values are under 1.0 x 10^{-9} Pa*m^3/s. In Figure 21, photographs of the cut section of the welded part through microscope, in which the indicative echoes were detected during UT inspections before drop test, are shown. Although a small air hole was observed, no crack initiation could be found in this figure. From these results, it seems that the occurrence of the crack initiation, may be avoidable for the drop events in the vertical orientation even with an impact load of more than 1,000 G was applied.

Seismic Test

It is preferred that the concrete cask be oriented vertically in the freestanding condition.[4] In order to evaluate the tipping-over phenomena under strong earthquake conditions, the excitation tests were performed with a scale-model concrete cask using a two-dimensional shaking table test, and the applicability of the energy spectrum approach was discussed.

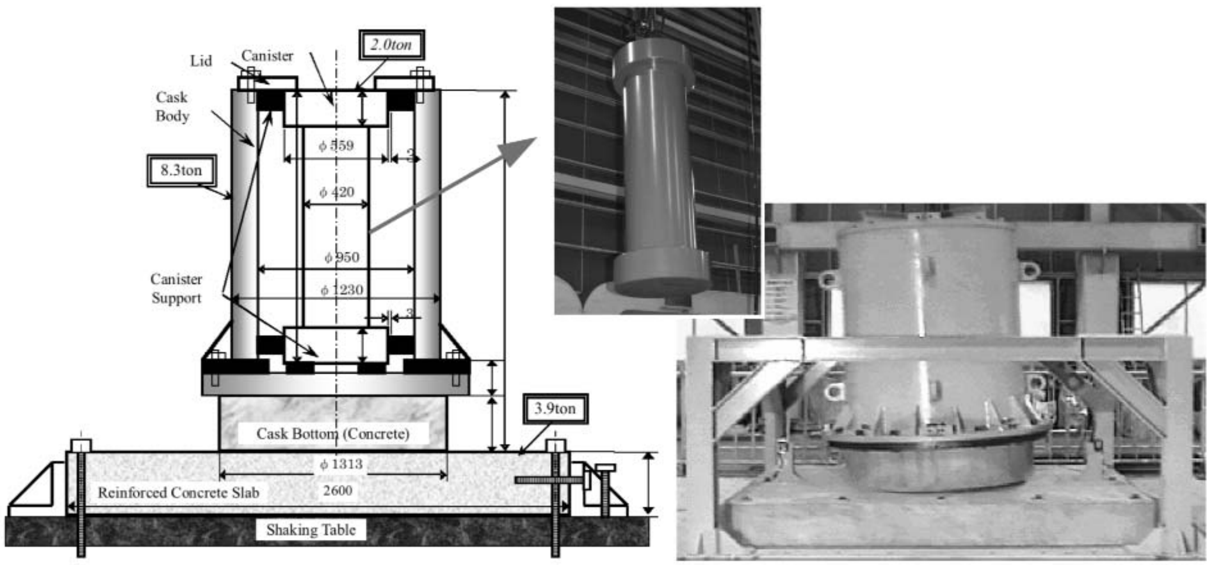
Scale-Model Cask

The scale-model cask and model floor were set on a two-dimensional (horizontal and vertical) shaking table as shown in Figure 22. A scale-model cask including the canister model was fabricated based on the similarity law referring the configuration of the RC type to simulate the effect of the gravitational acceleration on the tipping-over condition of the cask. The scaling ratios for acceleration, geometry, and bottom stress were set to 1, 1/3, 0.95, respectively. A 30 cm-thick reinforced concrete slab was used as the floor model. During the seismic excitation test, the angle, angular velocity, acceleration, and displacement of the cask body and the canister were measured.

Test Condition

For input of the seismic excitation test, waves were recorded during a typical natural earthquake waves and artificial seismic waves were employed. Duration of the input wave was scaled (1/1.73) according to a similarity law and the acceleration levels were varied according to the test conditions. Test condition includes the cases considering horizontal and vertical motion simultaneously. Moreover, the effect of the gap distance between the canister model and the cask body on the overall response of the scale-model cask was also investigated.

Figure 22. Scale-model cask



Test Results

Rocking Response

Before the test, the damping ratio for the rocking vibration and kinetic coefficient friction between the scale-model cask and the model floor were measured and set to 0.066 and 0.7, respectively. During the seismic response of the scale model, three-dimensional behavior such as top-spinning was observed. However, the residual sliding displacements were very small. Figure 23 shows the example of the test results using the wave recorded during Hyogo-ken Nanbu earthquake that occurred in 1995. The increase of maximum response angle by the effect of the vertical motion was up to 20 percent. It was also found that the gap between cask body and canister decreases the rotational angle response of the model cask.

Tipping-Over Criteria By Energy Spectrum

Akiyama et al.[5] proposed the estimation method for tipping-over the two-dimensional rigid rectangular body based on the energy spectrum approach. Figure 24 shows the model of the two-dimensional rigid rectangular body. If V_{Ereq} and ${}_{ou}V_E$ are defined as the equivalent velocity calculated from the critical potential energy of the rigid body and the equivalent velocity calculated from input energy to rigid body, respectively, the criteria for the tipping-over of the rigid body with energy spectrum is defined by Equation 1.

$$ {}_{ou}V_E(a) < V_{Ereq} = \sqrt{g(\sqrt{B^2 + H^2} - H)} \qquad (1) $$

$$ {}_{ou}V_{E\,modified}(a) - \sqrt{\int_{T_0}^{T_1} f(T)(V_{E0}(T))^2 dT} $$

$$ f(T) = -\frac{2(T - T_1)}{(T_1 \quad T_0)^2} $$

$$ T_0 = 0.02\sqrt{a} $$

$$ T_1 = 0.3\sqrt{a} $$

Figure 25 shows the maximum rotational angle and probability of tipping-over for energy spectrum. For the equivalent velocity of tipping-over limit (248 kine), the probability value is about 1.5 x 10^{-3}. Moreover both of the evaluated rotational angle and displacement are lower than the allowable value.

Conclusion

The demonstrative tests with the full-scale concrete cask had successfully concluded. The heat removal test using the full-scale RC and CFS type casks, drop tests using canisters, and the excitation tests using the 1/3 model have had been executed and evaluated.

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Figure 23. Maximum rotational angle response of a scale-model cask for excitation tests

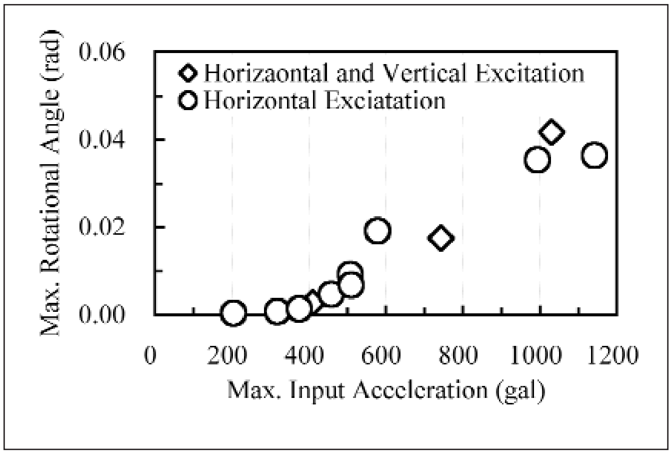


Figure 24. The model of the two-dimensional rigid rectangular body

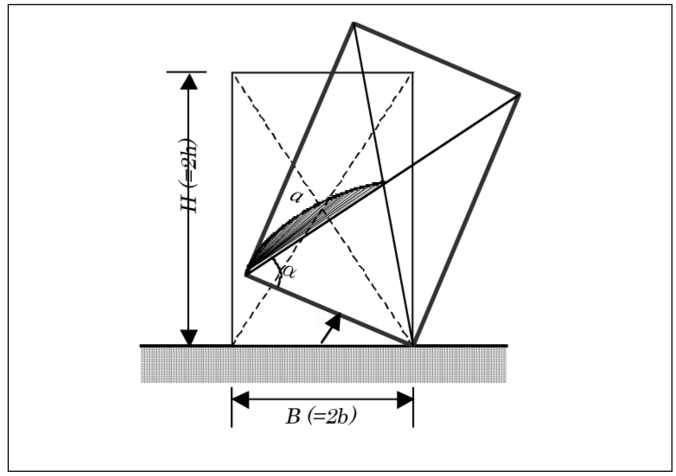
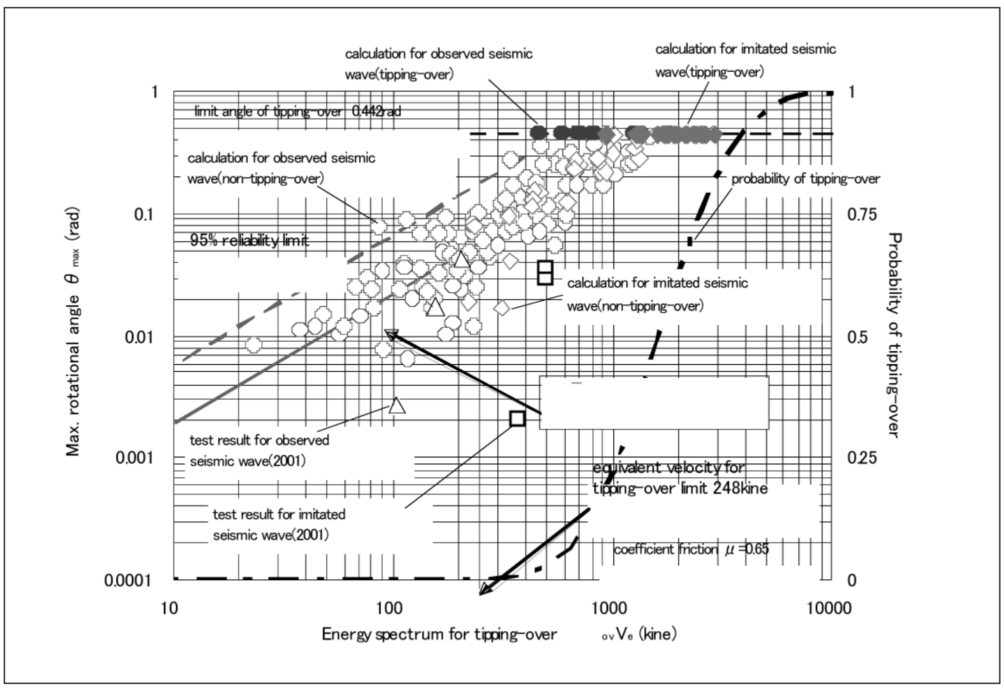


Figure 25. Maximum rotational angle and probability of tipping over for energy spectrum



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Transportation Licensing of Storage Canisters — Update and Lessons Learned

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Abstract

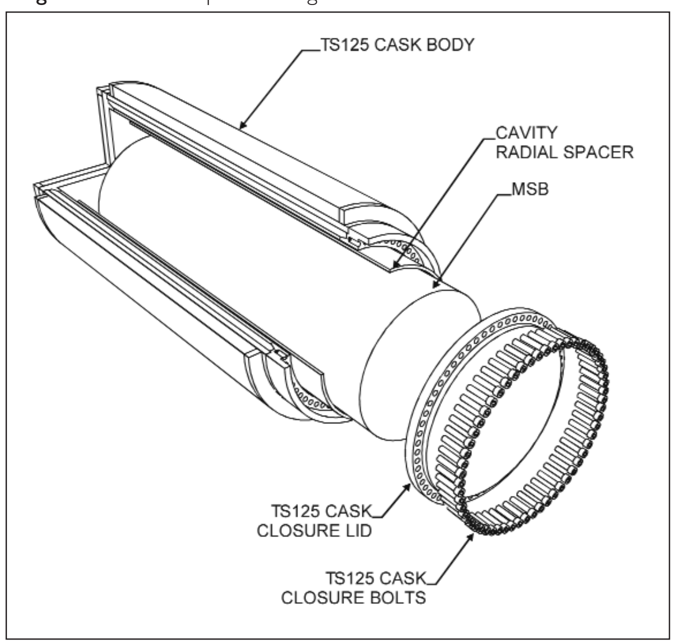
BNFL Fuel Solutions (BFS) is using a rigorous canister-specific burnup credit (BUC) evaluation approach to license the fifty-eight VSC-24 multi-assembly sealed baskets (MSBs) currently loaded and stored at three separate utility independent spent fuel storage installations (ISFSIs) for transportation in accordance with the requirements of 10 CFR Part 71. The canister-specific BUC evaluation approach rigorously models the specific characteristics of the spent nuclear fuel (SNF) assemblies in each MSB based on reactor records rather than the bounding design-basis SNF assembly characteristics typically used in generic BUC evaluations for transportation packages. It is anticipated that, by eliminating the conservative bounding assumptions typically used for a bounding generic BUC evaluation, many more of the MSBs will be shown to satisfy the transportation criticality acceptance criteria using the canister-specific BUC evaluation approach.

Preliminary canister-specific BUC evaluations show that many of the fifty-eight loaded MSBs satisfy the criticality acceptance criteria for transportation, based on current U.S. Nuclear Regulatory Commission (NRC) guidelines. However, in order to license the remainder of the MSBs for transportation, it is likely that a more aggressive BUC evaluation approach will be required. This could include BUC for fission products that do not currently have adequate benchmark data and/or a reduced administrative criticality safety margin. Based upon discussions with the NRC, application of more aggressive BUC criteria may be acceptable, given that the specific SNF assembly contents in the MSBs are well-characterized. However, because the more aggressive criteria reduce the margin of safety against criticality, those MSBs qualified using the more aggressive criticality acceptance criteria may also require a probabilistic risk analysis (PRA) to demonstrate that the probability of an accidental criticality event due to misloading under-burned or fresh fuel is sufficiently low. In this paper, we provide an overview of the canister-specific BUC evaluation approach to be used to license the fifty-eight existing loaded MSBs for transportation. In addition, we discuss a proposed PRA regulatory framework for those MSBs that require more aggressive criticality acceptance criteria.

Introduction

The VSC-24 Storage System (CofC No. 72-1007) is designed and licensed to safely store spent nuclear fuel (SNF) assemblies at an independent spent fuel storage installation (ISFSI) over a twenty-year storage period. The VSC-24 Storage System is composed of two primary components: a multi-assembly sealed basket (MSB) and a ventilated concrete cask (VCC). The MSB assembly includes an internal basket structure that positions and supports the SNF assemblies and a welded shell assembly that provides containment of radioactive materials. The SNF assemblies are stored in a dry inert environment inside the MSB to maintain their structural integrity. The MSB is stored inside the VCC, which provides natural convective cooling, radiation shielding, and physical protection from external events.

Figure 1. MSB Transport Package

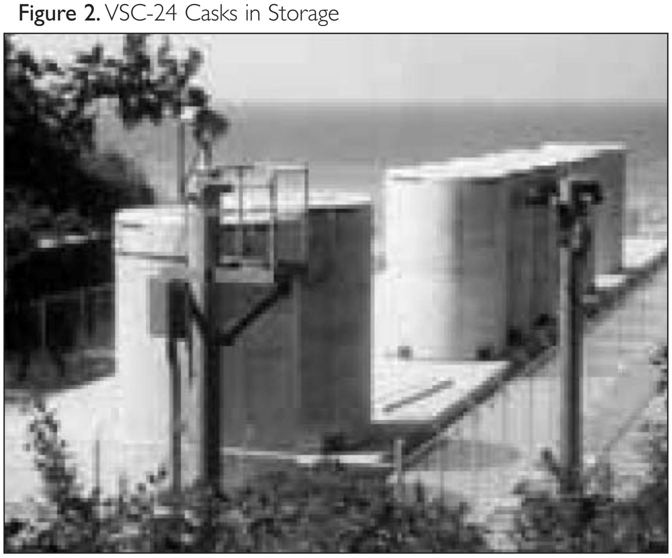


To date, a total of fifty-eight MSBs have been loaded and placed into storage at three different utility sites: Arkansas Nuclear One (ANO), Palisades, and Point Beach. At this time, all fifty-eight loaded MSBs must be opened and unloaded at end of the storage period in order to repackage the SNF assemblies for off-site transport to the Yucca Mountain repository. As an alternative, BNFL Fuel Solutions (BFS) is pursuing U.S. Nuclear Regulatory Commission (NRC) approval to transport the existing loaded MSBs directly to the Yucca Mountain repository using the NRC-certified FuelSolutions™ TS125 transportation cask and impact limiters (CofC No. 71-9276). An expanded view of the proposed transportation package is shown in Figure 1 (impact limiters not shown).

Background

From the mid-1980s to the early-1990s, most dry-cask storage systems included canisters that were designed and licensed for on-site storage conditions only. These storage canister designs typically do not include fixed poisons or flux traps in the baskets, but rely upon soluble boron credit for criticality control during fuel loading and unloading operations in the spent fuel pool. The storage canister shell assemblies are also typically designed to remain watertight under all design-basis normal, off-normal, and accident conditions; otherwise their use is limited to flood-free sites. The VSC-24 storage casks, shown in Figure 2, are typical of these first-generation systems.

Figure 2. VSC-24 Casks in Storage



The more recent dry-cask storage systems include dual-purpose canisters, designed and licensed for both on-site storage and off-site transportation. Unlike the earlier storage canisters, dual-purpose canister basket designs typically include fixed poisons, and flux traps in some cases, in order to maintain criticality control with fresh fuel and optimal moderation in accordance with the requirements of 10 CFR 71.55(b).

To date, more than 250 casks have been loaded with storage canisters that, under the current provisions, will have to be opened to repackage the SNF assemblies for shipment to the Yucca Mountain repository. The roughly 250 loaded storage canisters slated for repackaging include several different cask vendors' designs and are located at many different ISFSIs throughout the contiguous United States. Thus, the eventual disposition of the SNF assemblies in these storage canisters is a U.S. industry-wide issue. Most of the SNF assemblies loaded in storage canisters are old, cold fuel that is much less reactive than the design-basis spent fuel payload. Typical SNF assemblies loaded in these storage canisters have maximum assembly burnups in the 30,000 to 40,000 MWd/MTU range, maximum initial enrichments of approximately 3.0 to 3.5 wt percent U-235, and average assembly heat loads of approximately 0.5 kW. In many cases, the SNF payloads do not challenge the design-basis limits of the storage canister systems. Given the less reactive state of the specific SNF assembly payloads in the existing storage canisters, the canister-specific BUC evaluations will likely show that many of these existing storage canisters satisfy the criticality control requirements of §71.55(b).

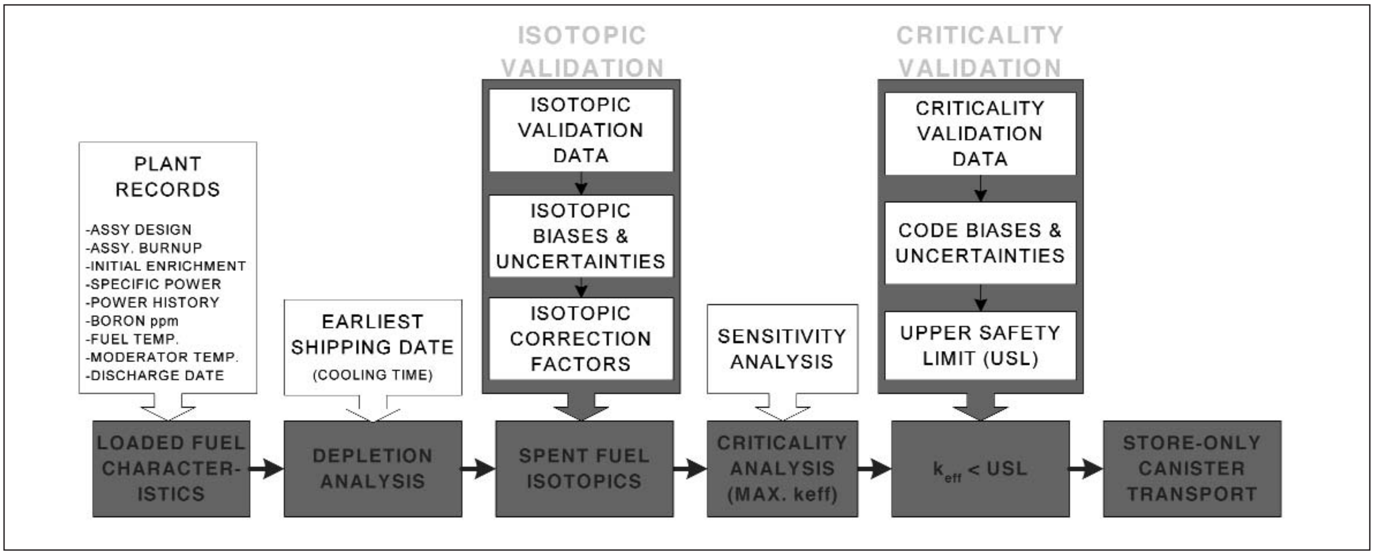
Transporting intact storage canisters, as opposed to unloading these canisters and repackaging the spent fuel for transportation, has many clear benefits. By transporting the loaded storage canisters directly to the repository, additional fuel handling operations at the plant are eliminated, which minimizes the probability of a fuel handling accident, has the least impact on on-going plant operations, and reduces occupational exposure for plant personnel. In addition to substantially reducing costs for the utilities and their ratepayers, the total infrastructure costs are also substantially reduced by performing most or all canister unloading and fuel repackaging operations at the repository facility.

Canister-Specific BUC Approach

BFS developed a canister-specific BUC approach that uses the known characteristics and burnup history of the SNF assemblies loaded in each MSB, based on reactor records, rather than bounding design-basis characteristics and burnup history parameters. The canister-specific BUC approach, which is much more rigorous than a typical BUC analysis for a generic transportation package design, provides a more accurate prediction of the reactivity of each specific canister payload and eliminates much of the unnecessary conservatism associated with bounding, design-basis assumptions.

A flowchart of the canister-specific BUC evaluation procedure is shown in Figure 3. The main steps of the canister-specific BUC evaluation procedure (i.e., fuel depletion analysis, criticality analysis, and isotopic and criticality validation) are very similar to

Figure 3. Flowchart of canister-specific BUC evaluation procedure



the generic BUC evaluation procedure. The main differences between the two procedures are the fuel depletion analysis inputs and the end-product of the BUC analysis. Whereas a generic BUC evaluation uses bounding SNF assembly parameters, minimum SNF assembly burnups for a given initial enrichment, and minimum cooling times to produces a fuel loading acceptance curve, the canister-specific BUC evaluation uses the actual materials and geometry, burnup histories, and cooling times for each specific SNF assembly loaded in an MSB to calculate the neutron multiplication factor (k_{eff}).

For the canister-specific BUC analysis, fuel depletion analyses are performed for each fuel assembly stored in an MSB using the SAS2H module of the SCALE 4.4 software package. The analyses are based on the specific fuel assembly geometry and materials (including burnable poisons), initial enrichment, discharge burnup, burnup history, and cooling time. SAS2H models are developed for each SNF assembly based on the specific fuel assembly geometry and assembly pitch within the reactor core.

The specific burnup history parameters and the related cooling time of each SNF assembly are used to calculate the isotopic concentrations. The following burnup history input parameters are used for each reactor cycle:

- Cycle start and end date
- Average cycle boron concentration
- Assembly average burnup at start of cycle
- Assembly average burnup at end of cycle
- Moderator temperatures (inlet and outlet)

A linear boron let-down function is modeled for each reactor cycle based on the average boron concentration during the cycle. The linear boron let-down function assumes that the boron is fully depleted at the end of each cycle.

The fuel assembly is modeled as eighteen individual, equally spaced axial segments to account for axial burnup variation effects. The local burnup of each axial region of the fuel assembly is calculated based on the assembly average end-of-cycle burnup and the bounding axial burnup profiles from DOE/RW-0472.[1] For each cycle, the local power of each axial zone is obtained by dividing the cycle-specific local burnup by the cycle duration.

The average moderator temperature at each of the eighteen axial regions of the fuel assembly is calculated using the reactor inlet and outlet moderator temperatures, based on the integration of a cosine power shape function. The temperatures of the fuel rods and fuel cladding at the mid-length of each axial zone are calculated based on the corresponding local moderator temperature and power. The cooling time for each assembly is determined based on the assembly discharge date and a specified earliest shipping date. Unlike a generic BUC analysis in which a minimum cooling time of five years is typically assumed, the canister-specific BUC analysis takes advantage of longer cooling times and the resulting decrease in reactivity primarily due to the decay of ^{241}Pu and buildup of ^{241}Am and ^{155}Gd.

A canister-specific criticality analysis is performed for each MSB and its specific SNF payload using the MCNP5 computer code. The isotopic concentrations in each of the 18 axial zones of all 24 SNF assemblies in an MSB are modeled using the results of the fuel depletion analysis. The isotopic concentrations are multiplied by isotopic correction factors to account for the bias and uncertainty of the SAS2H code; isotopes with positive reactivity are increased and those with negative reactivity are decreased. The transportation package is modeled in its most reactive credible configuration with optimum water moderation in accordance with the requirements of §71.55(b). The most reactive credible configuration is determined based on criticality sensitivity anal ses for the fuel types stored in the MSB, which examine the effects of assembly position, geometric tolerances, and moderation density on reactivity. The criticality analysis is performed to determine the neutron multiplication factor (k_{eff}) of the package. The calculated k_{eff} value for the package is compared to the upper safety limit (USL), which accounts for administrative margin and criticality code biases and uncertainties. Any biases or uncertainties of the criticality code are determined by validating the criticality code based on a series of applicable and representative critical benchmark experiments. The procedure used for the criticality code validation analysis is identical for both the canister-specific and generic BUC analysis procedures. In the canister-specific BUC analysis procedure, a canister is qualified for transport when its calculated k_{eff} is less than the corresponding USL.

PRA Regulatory Framework

The regulatory framework for criticality analyses of transportation packages consists of the applicable regulatory requirements and the analysis guidelines recommended by the NRC. In accordance with the requirements of §71.55, a transportation package is required to be designed and constructed and its contents limited such that, in its most reactive configuration under optimal water moderation, it remains subcritical. Current NRC guidelines also require a 5 percent administrative margin for criticality. Thus, the neutron multiplication factor (k_{eff}), when adjusted for code bias and uncertainty is limited to 0.95. Furthermore, current NRC guidelines allow BUC for actinide compositions only (i.e., actinide-only BUC). However, at least one cask vendor is currently requesting NRC approval of a transportation package based on limited-fission-product BUC.

A significant amount of conservatism is provided under the existing regulatory framework for criticality analyses of transportation packages. The conservatism inherent to this regulatory framework provides adequate safety margin against accidental criticality events resulting from fuel misloads or other unforeseen events, consistent with the NRC's defense-in-depth philosophy. Based on the results of preliminary MSB-specific BUC evaluations, some of the fifty-eight loaded MSBs are expected to satisfy the transportation criticality acceptance criteria when analyzed in accordance with the existing regulatory framework. However, many of the MSBs will require more aggressive approaches, such as reduced administrative margin and/or BUC for fission products having little or no benchmark data.

In recent meetings between BFS and NRC staff, it was discussed that those MSBs requiring a more aggressive criticality approach could be licensed for transportation, provided that they are accompanied by a PRA demonstrating that the risk of an accidental criticality event due to uncertainties in fuel burnups or fuel misloads (i.e., loading under-burned fuel) is not credible. BFS has developed a proposed regulatory framework for performing such a PRA.

The basic elements of the proposed PRA approach are similar to those recommended for dry-cask storage PRA.[2] However, unlike a traditional PRA in which consequences (doses) resulting from design-basis and beyond-design-basis accidents are quantified, the proposed PRA evaluates the probability of human errors associated with the plant burnup records and fuel misloads and the corresponding effects on system reactivity margin, similar to an approach used by EPRI to establish a technical basis for spent fuel burnup verification.[3] The main steps of the proposed PRA framework are:

- Identify initiating events
- Develop accident sequences (fault trees)
- Perform human reliability analysis
- Perform technical evaluations
- Perform quantitative PRA evaluation

The first step of the PRA is to identify the initiating events (e.g., human errors) that could result in the loading of fuel with non-conservative burnups and potentially lead to the criticality safety limit being exceeded. To identify the initiating events, information that may be pertinent to the accuracy of the plant burnup records or the probability of fuel misloads must first be gathered and reviewed. This information includes, but is not limited to, the generic and cask-specific procedural information for determining and verifying the fuel assembly burnup, managing and manipulating plant data related to burnup, and loading fuel into casks. Based on this information, the possibilities for human errors within the process that could result in the non-conservative misloads of fuel (i.e., under-burned fuel) are identified.

The next step of the PRA is to develop accident sequences that can significantly challenge critical safety functions and result in unacceptable end states. In the proposed PRA framework, criticality control is the only critical safety function considered and exceeding a criticality safety limit is the unacceptable end state. Event trees, which outline the possible sequences of events leading from an initiating event to the end state, can be used to depict and quantify the accident sequences. In our case, the initiating event could be viewed simply as a fuel misload. Alternatively, the initiating event could be the initial loading of the fresh fuel assembly into the reactor core, in which case the accident sequence would model the various possibilities of human errors that could lead to a fuel misload, with the possibility of exceeding a criticality safety limit.

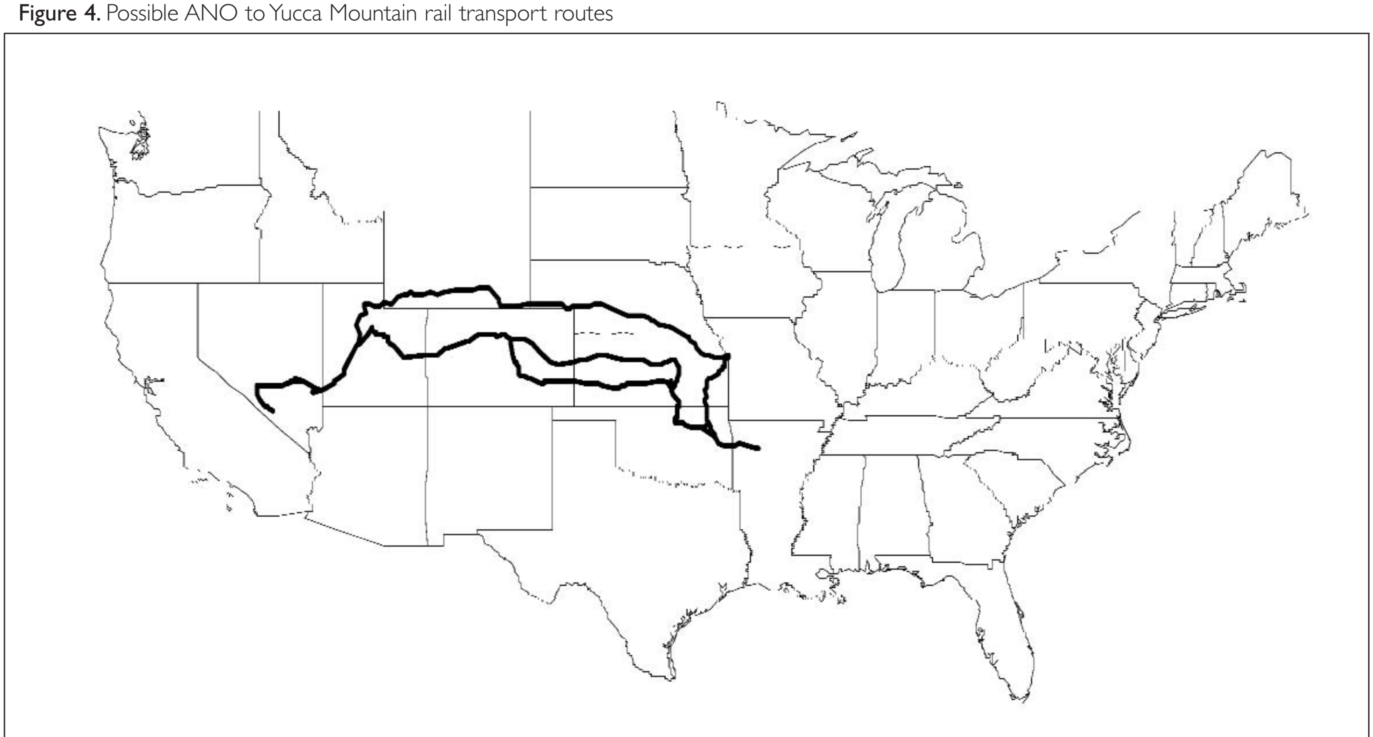
After the accident sequences are identified, a human reliability analysis (HRA) is performed to identify the error probabilities associated with the human errors that lead to the fuel misloads. The HRA may be performed using existing methodologies for at-power PRAs and dry-cask PRAs, and supported by information gathered from the plants. The human error probabilities may be based on published data or expert judgment. Using the individual human error probabilities, the overall accident sequence probabilities associated with the various types and extents of fuel misloads are determined. However, the probabilities of individual human errors must be combined in a manner that reflects whether they are dependent or independent failures. Criticality evaluations must then be performed to determine the effect of credible fuel misloads on package reactivity. These consist of criticality sensitivity evaluations of specific MSB payloads, considering potential misloads of underburned fuel or fresh fuel. These analyses are used to determine the number of underburned or fresh fuel assemblies that must be loaded into a specific MSB to cause the criticality safety limit to be exceeded for the more-aggressive BUC approach. Then the probability of the number of misloads required to exceed the criticality safety limit (i.e., criticality misload probability) can be determined based on the overall accident sequence probabilities and dependence factors determined in the HRA.

Other PRAs have shown that the probability of a non-conservative misload is between 10^{-3} and 10^{-5} for large casks qualified using a generic BUC evaluation approach.[4] For the canister-specific BUC evaluation, the probability of a fuel misload is expected to be lower than those for a generic BUC cask, given that the fuel has already been loaded into the MSBs and a significant number of burnup measurements have been taken to confirm the burnup records. Because the criticality safety margin associated with the more aggressive BUC approach is lower than that for the generic BUC approach, fewer non-conservative fuel misloads may be required for the MSB to exceed the criticality safety limit. However, it may be possible to show on the basis of the misload probabilities alone, that the number of non-conservative misloads required to exceed the criticality safety limit, conservatively assuming that the probability of optimal water moderation is 1.0 (i.e., a regulatory requirement), is not credible (i.e., probability less than 10^{-6}).

Since optimal fresh water moderator must be present to achieve the neutron multiplication factors determined in the criticality analyses, the PRA should account for the probability that a rail transport accident severe enough to breach the package containment system and allow water ingress will occur during MSB transport. The associated rail accident probability should be multiplied by the critical misload probability to obtain the overall probability that an accidental criticality event could occur during transport of the MSB.

Recent NRC-sponsored studies[5] have shown that the accident rate for rail transport in the United States is approximately 0.11×10^{-6} accidents/railcar-km, and that only 0.16 percent of all rail accidents severe enough to damage the containment system occur over water. Thus, the probability of a rail accident that results in water ingress is approximately 1.76×10^{-10} accidents/railcar-km traveled. The PRA is based on specific transport routes determined using current transport routing codes, such as TRAGIS.[6] For instance, the maximum distance traveled from ANO to the Yucca Mountain repository, considering the three alternative routes shown in Figure 4, is approximately 3,500 kilometers (2,200 miles). Thus, the probability of a rail accident

Figure 4. Possible ANO to Yucca Mountain rail transport routes



occurring during transport of an ANO MSB to Yucca Mountain that results in water ingress is approximately 6.2 x 10^{-7}. When this value is multiplied by the critical misload probability, the overall probability of an accidental criticality event occurring during transport is obtained.

Conclusions

The eventual disposition of the spent fuel assemblies loaded in canisters currently designed and licensed only for on-site storage conditions is an industry-wide issue. The canister-specific BUC evaluation approach developed by BFS can be used to license many of these storage canisters for transportation. This will allow these storage canisters to be transported intact to Yucca Mountain, thereby minimizing fuel handling operations, impact on plant operations, and occupational exposure, as well as total infrastructure costs.

Transportation certification for storage canisters having more reactive spent fuel payloads may require reliance on BUC approaches that are more aggressive than current NRC guidelines allow, such as credit for fission products for which sufficient chemical assay data is not available or reduction of criticality safety margins. For these more-aggressive BUC approaches, BFS has developed a probabilistic risk assessment (PRA) framework that can be used to support the NRC-approval basis. The PRA framework evaluates the probability that fuel misloads, caused by human error, could result in an accidental criticality event during transport.

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Safeguarding Geological Repositories for Spent Nuclear Fuel: Past, Present, and Future

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Introduction

Under the Nonproliferation Treaty (NPT) a non-nuclear weapons state signatory of the treaty must conclude a comprehensive safeguards agreement with the International Atomic Energy Agency (IAEA) (based on the INFCIRC/153 model agreement). The *traditional* safeguards system, based on the implementation of the safeguards agreement, has been applied to many types of nuclear facilities like reactors, critical facilities, conversion and fuel fabrication plants, separate storage installations, reprocessing plants, and enrichment plants. This safeguards system provides credible assurance of the non-diversion of nuclear material from declared nuclear activities. Upon the introduction of the Model Additional Protocol (INFCIRC/540 Corr.), to strengthen IAEA safeguards, the safeguards objective has been expanded to provide, as well as assurance of non-diversion, credible assurance of the absence of undeclared nuclear material and activities in the state.

The optimal combination of safeguards measures under a safeguards agreement and an additional protocol, called integrated safeguards, is at present being developed and implemented by the IAEA and its member states.

A new type of nuclear facility that will require implementation of safeguards in the near future is the geological repository. A geological repository is an underground installation for the disposal of radioactive waste, including nuclear material, e.g., spent fuel, usually located several hundred meters below the surface in a stable geological formation that ensures long-term isolation of radionuclides from the biosphere.

No facility of this type has ever been under IAEA safeguards and therefore the agency has no practical experience with them. Nevertheless a substantial amount of work has been invested to develop an approach for safeguarding this type of facility. This has been done mostly following the premises of traditional safeguards.

The SAGOR Project

The early interest in developing IAEA safeguards at geological repositories goes back to 1988 when an Advisory Group Meeting, gathering representatives of eighteen states, was called by the IAEA and made by consensus the important recommendation that safeguards should not be terminated on spent fuel after it has been emplaced in a repository and the repository has been closed.

In 1994, as an outcome of a 1991 Consultants Meeting on concepts for safeguarding geological repositories, the IAEA started the Program for the Development of Safeguards for the Final Disposal of Spent Fuel in Geological Repositories (SAGOR). The objective of SAGOR was to develop a generic safeguards approach that would provide the IAEA with a reference for site specific approaches to verify the non-diversion of spent fuel from geological repositories. A second consultants meeting complemented and contributed to the development of the SAGOR project. After almost four years of continuous work, in December 1997, a second advisory group meeting endorsed the generic safeguards approach developed by SAGOR

SAGOR used as a reference model a geological repository formed by tunnels and shafts located in a geological formation of salt, clay, tuff, or crystalline rock at depths between 250 and 1,000 meters. Containers (casks or canisters in shielded overpacks) would be filled with spent fuel or waste containing nuclear material at a surface conditioning/encapsulation facility and subsequently transported to the repository tunnels where they would be emplaced for final disposal. The spent fuel in a filled repository was expected to contain 2,000 to 200,000 metric tons of uranium and 10 to 2,000 metric tons of plutonium. The operational life of a repository was expected to be twenty to one-hundred years before backfilling and closure.

SAGOR recognized that because of the unique characteristics of geological repositories, safeguards implementation would present some problems that would make monitoring and verifying repositories difficult. For example:

- The integrity of the geological formation into which the repository will be constructed will be only partially or indirectly observable.
- Construction activities in the repository may be carried out simultaneously to emplacement of spent fuel, complicating verification and monitoring.
- The canisters containing spent fuel should not to be reopened and the continued presence of emplaced containers in backfilled or sealed tunnels could not be directly verified, if the design safety criteria of the repository are maintained.
- A closed repository could be re-excavated or accessed, e.g., by undeclared tunnels. The difficulty would mainly depend on the type of rock constituting the geological formation.

Excavation could proceed faster and at lower costs as technology improves with time.
- Repositories will be built and operated over decades and the post-closure phase would cover thousands of years.

The monitoring objectives of a geological repository for safety and safeguards purposes run in parallel. Safety measures must ensure that the radionuclides emplaced in the geological repository remain within the perimeter of the repository until they decay to safe levels. Safeguards must assure that the nuclear material is emplaced and detect undeclared activities intent on recovering it. But safety considerations must take precedence and reverification activities are to be designed and implemented to ensure that the radiological safety of the repository is not jeopardized. The safeguards approach will therefore not require reverification of the nuclear material once it has been transported underground and definitely exclude reverification of the nuclear material contained in a closed repository.

Based on these and other considerations, SAGOR developed a generic safeguards approach applicable to the model repository. At the time it was developed, this safeguards approach was mainly based on traditional safeguards considerations as derived from INFCIRC/153(Corrected) and the safeguards implementation experience at other facility types.

SAGOR defined three phases in the life of a geological repository:

- Pre-operational phase: defined to be the period from designation by a state of a site for construction of a geological repository up to reception of the first disposal container. It may include site characterization, underground exploration, and construction of the repository. In some states, site characterization (including access construction and underground exploration) may occur before national designation of the geological repository site.
- Operational phase: begins when the first disposal container arrives and continues through final closure of the repository. It includes receipt of disposal containers, tunnel and vault excavation, emplacement of containers, deployment of engineered barriers, tunnel and vault backfilling, and repository backfilling and sealing. Various barriers and seals may be put in place at different times depending on waste and rock characteristics. Requirements for waste package retrievability may have a significant influence on the disposal options chosen.
- Post-closure phase: starts when the geological repository has been backfilled and closed and the surface facilities have been decommissioned. Closure is defined as a permanent condition with respect to the status of the geological repository at the end of its operating life. Monitoring and surveillance for safety purposes could be maintained for as long as society considers it beneficial, although it is a design principle of geological disposal that assurance of safety should not require post-closure monitoring.

The SAGOR geological repository generic safeguards approach is based on maintaining knowledge of the quantity of spent fuel placed into the geological repository. Success of the safeguards approach is dependent on confirming the integrity of the geological containment, verifying that spent fuel containers are not removed undeclared through any pre-existing or unknown opening into the underground facility, verifying that spent fuel containers cannot be converted into smaller items whose removal would be more difficult to detect, and verifying that the spent fuel is not processed to recover the nuclear material content.

At the spent fuel conditioning/encapsulation plant located on the surface, the safeguards approach is based on nuclear material accountancy, and containment and surveillance (C/S). The safeguards approach may use techniques such as nondestructive assay, seals, surveillance cameras, motion sensors, radiation detectors, and weight monitors. These measures would be applied in both the cask handling areas and at the hot cells for manipulating fuel. If the spent fuel was previously verified before shipment to the encapsulation/conditioning or surface reception plant and continuity of knowledge has been maintained further up to the encapsulation/conditioning/reception plant and, additionally, there is assurance that the disposal container leaves this facility under effective containment and surveillance until final emplacement, C/S could be the main safeguards measure to be applied.

In terms of the repository itself, the primary safeguards measure below ground would be design information verification, with additional measures applied, if needed. Design information verification conducted below ground will include visual observation and possibly geophysical techniques (e.g., active and passive seismic, active and passive electromagnetic, ground-penetrating radar) when the characteristics of the geological formation allows for their application. Environmental sampling through air sampling and samples for trace analysis will be used to detect opening of casks and underground reprocessing of spent fuel.

With respect to the above ground areas inspector direct observation, and aerial and satellite surveillance activities will play also an important role for verification of the surface area inside a reasonable radius around the repository location. Radiation monitors and C/S measures may be applied to ventilation and access shafts at the surface level to detect and verify transfers of radioactive containers into and out of the repository.

After the geological repository has been backfilled closed, the safeguards approach contemplates maintaining knowledge of the spent fuel content of the repository through the use of satellite and unattended geophysical monitoring.

SAGOR covered an important phase in the development of a safeguards approach for geological repositories. Upon delivery of the SAGOR report a Geological Repository Safeguards Experts Group was formed to continue providing recommendations and expertise to the IAEA in respect to repository safeguards. It operated under the umbrella of a joint task force of the Member States Support Program to IAEA Safeguards (MSSP).

IAEA Safeguards Policy on Geological Repositories

Based on the SAGOR generic approach and recommendations made by the Expert Group the IAEA Department of Safeguards issued in June 1997 a Policy Paper[1] providing guidelines for safeguarding geological repositories. The following are some concepts contained in this policy paper:

- Spent fuel disposed in geological repositories is subject to safeguards in accordance with the IAEA's safeguards agreement with the state. Safeguards on the spent fuel will be maintained after the repository has been backfilled and sealed, for as long as the safeguards agreement remains in force.
- The safeguards system will be based on (a) verification of the repository design information during design, construction, and operation; (b) verification of receipts and flows to ensure that no undeclared removal of nuclear material occurs; and (c) maintenance of continuity of knowledge on the nuclear material content.
- Safeguards requirements should be integrated into the repository design at an early stage to establish functional, non-intrusive, and cost-effective safeguards measures.
- Because emplaced spent fuel cannot be reverified, sufficient redundancy, diversity, and robustness should be incorporated into the safeguards approach and adequate maintenance measures should be applied to avoid system failure and to ensure continuity of knowledge.
- The safeguards measures should be designed to provide verification of the nuclear material contents of incoming spent fuel containers and continuity of knowledge of the spent fuel inventory. As far as practical, the safeguards measures should function in automated, remote control, and remote data transmission modes.
- Design information verification should confirm the design of the geological repository, including its surface facilities, and detect any undeclared activities, both in the repository and in its vicinity. The excavation areas should be verified to be as declared and to ascertain there are no undeclared excavations. The safeguards approach should provide assurance of the absence of undeclared underground reprocessing and assurance of no undeclared operational capability underground that could facilitate and mask transfer of spent fuel to other containers.
- In the pre-operational phase, the IAEA, in collaboration with the state, should establish the baseline information: all safeguards-pertinent information about the original undisturbed site. Design information verification should be conducted on all excavations and facilities.
- During the operational phase of the geological repository, the safeguards system should verify transfers, flows, and inventory of spent fuel disposal containers through the application of elements of containment and surveillance, monitoring, non-destructive assay, and design information verification.
- The safeguards measures should verify that the containers declared to be emplaced are actually emplaced; that the containers are not opened, switched, or removed; and that there are no undeclared operations that could change the amount of nuclear material in any disposal container.
- During the post-closure phase, the safeguards system should consist of sufficient surface monitoring measures to provide assurance of no undeclared attempts to obtain access to nuclear material. These measures may include information analysis, visual observation, remote surveillance (e.g., satellite), geophysical monitoring, and environmental techniques.

This policy paper constitutes as of today the main guideline to direct safeguards implementation for those geological repository projects under development around the world.

Status of Geological Repository Projects and Present Safeguards Activities

Facilities for the final disposal of spent fuel are at various stages of planning, scheduling and approval. Site characterization activities are being conducted and some exploratory facilities have been excavated. The IAEA is confronted with the need to take the first steps to prepare for the future implementation of safeguards at these facilities.

- The United States designated the Yucca Mountain site as its repository in July 2002. The exploratory facility at Yucca Mountain is complete and will become part of the repository. As of the end of 2004 the U.S. repository is scheduled to begin operation in 2010.
- Sweden is expected to designate its final repository location in early 2008. The Swedish repository is scheduled to start operating in 2015.
- Finland designated in May 2001 the Olkiluoto site for its national geological repository. Finland's underground laboratory, Onkalo, whose construction began in July 2004, would become part of its repository upon successful completion of rock characterization. The repository is scheduled to start operating in 2020.
- In Germany, completion of site selection actions has been postponed for three to ten years under a government moratorium.
- Belgium, Switzerland, France, Germany, Canada, and Sweden have excavated underground laboratories to confirm their repository concepts.

Geological repository programs in many countries (e.g., Canada, Belgium, Switzerland, Hungary, Czech Republic, Spain, France, and UK) are either delayed further into the future or the countries have not yet officially declared their policies or schedules. Repository conceptual designs have nevertheless been prepared in several countries and some will be submitted for licensing consideration in the next years.

Against this background of activity by IAEA member states in the development of geological repositories, the Geological Repository Safeguards Experts Group, in which all states with active repository projects have designated experts, continued its activities through its annual meetings and the interaction of its members. For example, an experts' meeting in 2002 evaluated the areas in which monitoring needs of the IAEA, state regulatory authorities and the repository operator may overlap. Verifying the integrity of geological containment, assuring conformance with approved designs, verifying cask management and oversight, detecting movements of materials, verifying operational activities, and detecting radiological events were identified areas. The group has been also highly instrumental in maintaining the flow of information between member states and the IAEA and in determining needs for developing research tasks related to geological repositories.

Independent of the experts group activities, an IAEA Waste Technology Consultants' Meeting in 2003 assessed the technological implications of IAEA safeguards on geological repositories and concluded that the impact of IAEA safeguards will not be significantly different from that at other nuclear facilities. Implementation of the safeguards measures would be consistent with envisioned operational controls and management practices and all IAEA safeguards impacts were expected to be resolvable through advance planning and cooperation.

Present activities in terms of safeguards implementation for geological repositories involve the examination and verification of baseline and design information and the development of safeguards measures and techniques specifically related to the existing repository projects. The goal is to implement the applicable parts of the IAEA safeguards policy and, based on the generic safeguards approach and relevant new developments, to design specific safeguards approaches for the geological repositories to be constructed and operated in the near future.

The implementation of IAEA safeguards measures for the pre-operational phase of a geological repository is starting to be applied at Finland's Olkiluoto Geological Repository. Finland has officially informed the IAEA of the Olkiluoto repository project and the excavation of the related underground laboratory, Onkalo. The IAEA and Finland have met to discuss technical steps to be taken to facilitate the implementation of safeguards for Finland's repository, beginning with baseline and design information provision and design information verification measures.

Also several research and development support program tasks (under the MSSP) directly related to the safeguards measures applicable to Olkiluoto are being proposed. They will cover feasibility studies and testing of particular satellite sensors (radar interferometry, hyper spectral) applicable to the peculiar conditions of this repository located at a northern latitude (snow, clouds, winter darkness). Other tasks will investigate the potential use and constraints of passive seismic monitoring techniques and ground penetrating radar, in the particular conditions of the Olkiluoto rock formation.

The Future Development and Implementation of Safeguards at Geological Repositories

A number of issues will have to be clarified or further developed in the near future. For example:

- Methods to evaluate the baseline information available during the pre-operational phase of a geological repository with the purpose to establish sufficient certainty on the initial integrity of the geological formation
- The structure and content of design information and of design information questionnaires (DIQ) applicable to the different phases of geological repositories
- Techniques to perform design information verification (DIV) at the various phases of repositories, in particular for the active projects taking into consideration their particularities
- Potential problems associated with application of safeguards at mixed repositories containing spent fuel and other radioactive waste not subject to IAEA safeguards

Another important area that the IAEA and its advisory bodies have to investigate is the impact of the measures contained in the Model Additional Protocol in relation to the generic safeguards approach for geological repositories. As mentioned before this generic approach was developed mainly taking into consideration traditional safeguards concepts based only on the implementation of the safeguards agreement. A quick examination of the list of states involved in repository projects, present and future, reveals that they will have additional protocols in force and most will be under integrated safeguards before their repository begins operations. How this will affect the particular safeguards approach for their repositories is still an open question. In that respect it is also important to periodically re-examine the applicability of Policy Paper No.15 and its validity to support safeguards implementation at active repository projects and to make modifications, if required.

At a recent meeting, in December 2004, of representatives of the states participating in the Geological Repository Safeguards Experts Group the participants decided that this group had facilitated the sharing of information between member states and the Agency and performed an important advisory function to the IAEA in respect to the development of safeguards approaches and techniques applicable to repositories. The participants recommended the continuation of the work of the group, under a renovated terms of reference. The work of the renovated group will be now mainly focused on practical aspects of the application of the generic safeguards approach and related techniques to specific geological repository sites. The new name of the group will be "Application of Safeguards to Geological Repositories, Group of Experts" (its acronym will be "ASTOR"). The duration of the activities of this new group will extend across the next ten years and an invitation to participate in the group will be extended to a larger number of member states. Inter allia it will advise and support the IAEA to:

- Facilitate sharing of information on geological repositories between member states
- Help promote understanding of safeguards requirements and implications in other technical forums and with operators of geological repositories
- Supply experts and expertise for safeguards missions and implementation support related to repositories and
- Study and make recommendations on any issues that may arise regarding geological repository safeguards, including those mentioned above

Safeguards activities carried out by the IAEA Department of Safeguards increasingly will be implemented at the site of the Olkiluoto repository in cooperation with the Finnish authorities and operators. In relation with the Swedish repository, activities are foreseen to start after the site is designated and the Swedish government informs the Agency. With respect to Yucca Mountain any safeguards activities will depend on the voluntary offering of the U.S. government according to its safeguards agreement with the IAEA. In any case further U.S. support in the development and testing of specific techniques and instruments will be welcomed.

End Notes

1. Report of the Advisory Group Meeting on Safeguards for Final Disposal of Spent Fuel in Geological Repositories (AGM-995), STR-309, International Atomic Energy Agency, December 1997.
2. Policy Paper 15: Safeguards for Final Disposal of Spent Fuel in Geological Repositories. Safeguards Manual, Department of Safeguards, IAEA.
3. Report of the Advisory Group Meeting on Safeguards for Final Disposal of Spent Fuel in Geological Repositories (AGM-995), STR-309, International Atomic Energy Agency, December 1997.

4 Policy Paper 15: Safeguards for Final Disposal of Spent Fuel in Geological Repositories. Safeguards Manual, Department of Safeguards, IAEA.

U.S. and UK Ink Contracts on Idaho and East Tennessee Projects

The U.S. Department of Energy (DOE) and the United Kingdom's Department of Trade and Industry (DTI) announced in February the execution of two contract modifications providing for the DOE's accelerated purchase of the Advanced Mixed Waste Treatment Project (AMWTP) located at the Idaho National Laboratory (INL) and the resolution of contract claims associated with work conducted by BNFL at the AMWTP site and the East Tennessee Technology Park (ETTP) in Tennessee.

BNFL constructed the AMWTP under a fixed-price contract awarded in 1996. BNFL has been decontaminating and decommissioning three buildings at the ETTP under a fixed-price contract awarded in 1997.

The AMWTP is a set of facilities constructed by BNFL with the capability to sort, segregate, volume reduce, and repackage waste in order to facilitate its final disposal. Under the original AMWTP contract, DOE would have paid BNFL $568.3 million in a series of installments over a period of six to eight years to amortize the construction and start-up cost of the facilities, at which time title to the AMWTP facilities would have transferred to DOE.

Additionally, under the original contract BNFL would have been paid on a unit rate basis to process waste through the AMWTP facilities. Under the AMWTP contract modification, DOE will pay $428 million for the AMWTP facilities and receive title to the facilities in early May 2005, after the facilities are fully operational and satisfactorily complete a certification audit, scheduled for late February 2005.

The work at ETTP is expected to be completed in May 2005, after which consideration may be given to making some of the cleaned up facilities available for reindustrialization activities.

The total value of the two contract modifications, consisting of the purchase price for the AMWTP facilities, resolution of outstanding claims under both contracts, and incentive payments, will be between $500 million and $550 million (plus previously established remaining payments of approximately $16 million for the ETTP project completion).

DOE Exercises Option on Washington TRU Solutions WIPP Contract

The U.S. Department of Energy (DOE) announced in February that it will exercise the five-year option in the Washington TRU Solutions LLC (WTS) contract to continue managing and operating the DOE's Waste Isolation Pilot Plant (WIPP) in New Mexico. The WTS contract will now run through September 30, 2010, with an estimated value over the five years of approximately $704 million.

New performance goals have been set to be completed between 2006 and 2010, including:

- Disposal of 20,000 additional cubic meters of TRU waste in the WIPP underground repository
- Completion of the TRU waste cleanup at some storage sites five to ten years early
- Cleanup 70 percent of all legacy TRU waste by the end of 2010, compared to 53 percent targeted during prior planning

WIPP is the world's first underground repository certified to safely and permanently dispose of transuranic radioactive waste from the research and production of nuclear weapons and is a cornerstone of DOE's environmental cleanup program.

DOE Awards Portsmouth Remediation Contract to LATA/Parallax

The U.S. Department of Energy (DOE) has awarded a $141,261,897 small business contract to perform environmental remediation and waste management activities at the department's Portsmouth Gaseous Diffusion Plant in Piketon, Ohio, to LATA-Parallax Portsmouth LLC, a small business joint venture between Los Alamos Technical Associates Inc. and Parallax Inc.

The contract will run through September 30, 2009, and provides incentives to the contractor for managing costs effectively while completing the cleanup work. A second small business contract for infrastructure and maintenance activities at Portsmouth is currently in the procurement process.

LATA-Parallax will be responsible for groundwater and soil remedial actions, removing legacy waste, decontamination and decommissioning facilities, highly enriched uranium disposition, operating the site waste storage facilities, and surveillance and maintenance activities, as well as other activities. The group takes over from Bechtel Jacobs Company LLC, whose contract expires March 31, 2005.

LATA-Parallax is owned by Los Alamos Technical Associates Inc. (LATA), a New Mexico-based engineering, environmental, and nuclear operations services company, and Parallax Inc., a Maryland-based engineering, environmental, and nuclear operations services company. LATA is a service-disabled veteran owned small business and Parallax is a women-owned, minority-owned small business.

The Portsmouth/Paducah Project Office (PPPO) will manage three major contractors at the Portsmouth site under the environmental management cleanup mission. This office is responsible for the LATA-Parallax contract, a yet-to-be-awarded infrastructure contract, and the ongoing work of Uranium Disposition Services LLC, which is responsible for the Depleted Uranium Hexafluoride Conversion Project. Opportunities will likely be available for various subcontracts awarded by the major contractors for specific tasks. In addition, PPPO also oversees several activities conducted by the United States Enrichment Corporation.

IAEA/OCED Host Conference on Nuclear Power for the 21st Centruy

Ministers from twenty-nine countries have confirmed they will attend a two-day conference in Paris, March 21-22, "Nuclear Power for the 21st Century," organized by the International Atomic Energy Agency and hosted by the government of France in cooperation with the Organization for Economic Cooperation and Development (OECD) and the OECD Nuclear Energy Agency (NEA).

The ministers, with government officials from an additional thirty countries, will examine the future role of nuclear energy in meeting the energy needs of the world and present their views on the current and future role of nuclear power in the context of national energy strategies. Discussions at the conference will cover issues such as world energy needs and resources, environmental challenges, energy choices and governance, including compliance with non-proliferation undertakings.

Further details on the conference are available at www.parisnuclear2005.org

World Nuclear University Selects First Fellows

More than seventy top young nuclear professionals and academics have gained fellowships to study at the World Nuclear University´s first annual Summer Institute, July 9-August 20, 2005, in the United States. About half of the fellows are from developing countries and countries in transition, with financial backing from the IAEA´s technical cooperation program. Women comprise one quarter of the fellows, who come from thirty-four countries.

WNU was founded in September 2003 by four Founding Supporters: the International Atomic Energy Agency (IAEA), the Nuclear Energy Agency (NEA) of the OECD, the World Association of Nuclear Operators (WANO), and the World Nuclear Association (WNA). The IAEA helped to shape the WNU Summer Institute´s six-week educational program, and participated in the selection of Fellows. The course will be held at Idaho National Laboratory with support from the U.S. Department of Energy.

For more information on the World Nuclear University see http://world-nuclear-university.org/.

Chu Resigns Civil Radioactive Waste Management Post

The U.S. Department of Energy announced that Dr. Margaret Chu, director of the Office of Civilian Radioactive Waste Management, resigned effective on or about February 25, 2005. On March 6, 2002, the U.S. Senate unanimously confirmed Dr. Chu as director of the Department of Energy's Office of Civilian Radioactive Waste Management. As director, Dr. Chu was responsible for advising the Secretary of Energy and the President on issues surrounding the ongoing scientific research and licensing of the nation's first permanent geologic repository for spent nuclear fuel and high-level radioactive waste.

Advertiser Index

Author Submission Guidelines

The *Journal of Nuclear Materials Management* is the official journal of the Institute of Nuclear Materials Management. It is a peer-reviewed, multidisciplinary journal that publishes articles on new developments, innovations, and trends in safeguards and management of nuclear materials. Specific areas of interest include physical protection, material control and accounting, waste management, transportation, nuclear nonproliferation/international safeguards, and arms control and verification. *JNMM* also publishes book reviews, letters to the editor, and editorials.

Submission of Manuscripts: *JNMM* reviews papers for publication with the understanding that the work was not previously published and is not being reviewed for publication elsewhere. Papers may be of any length.

Papers should be submitted in *triplicate*, **including a copy on computer diskette**. Files should be sent as Word or ASCII text files only. Graphic elements must be sent in TIFF format in separate electronic files. Submissions should be directed to:

Dennis Mangan
Technical Editor
Journal of Nuclear Materials Management
60 Revere Drive, Suite 500
Northbrook, IL 60062 USA

Papers are acknowledged upon receipt and are submitted promptly for review and evaluation. Generally, the author(s) is notified within sixty days of submission of the original paper whether the paper is accepted, rejected, or subject to revision.

Format: All papers must include:

- Author(s)' complete name, telephone and fax numbers, and e-mail address
- Name and address of the organization where the work was performed
- Abstract
- Camera-ready tables, figures, and photographs in TIFF format only
- Numbered references in the following format:
  1. Jones, F. T. and L. K. Chang. 1980. Article Title. *Journal* 47(No. 2): 112–118.
  2. Jones, F.T. 1976. *Title of Book*, New York: McMillan Publishing.
- Author(s) biography

Peer Review: Each paper is reviewed by at least one associate editor and by two or more reviewers. Papers are evaluated according to their relevance and significance to nuclear materials safeguards, degree to which they advance knowledge, quality of presentation, soundness of methodology, and appropriateness of conclusions.

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Calendar

April 17 – 21, 2005
Monte Carlo 2005: The Monte Carlo Method: Versatility Unbounded in a Dynamic Computing World
Chattanooga, Tennessee, U.S.A.
Sponsor:
American Nuclear Society
Web site: http://meetingsandconferences.com/MonteCarlo2005

April 25 – 28, 2005
Workshop on Safeguards and Security System Effectiveness
Augusta Towers
Augusta, Georgia, U.S.A.
Sponsors:
INMM Materials, Control, and Accountability Technical Division and the INMM Physical Protection Technical Division
Contact:
Ed Sadowski
Phone: 803/952-2460
E-mail: edward.sadowski@srs.gov

May 8 – 11, 2005
Waste Management, Decommissioning and Environmental Restoration for Canada's Nuclear Activities
Crowne Plaza Hotel, Ottawa, Ontario, Canada
Sponsor:
Canadian Nuclear Society
Contact:
Canadian Nuclear Society
E-mail: cns-snc@0n.aibn.com
Web site: http://www.cns-snc.ca/waste_05.html

May 16 – 20, 2005
Third Russian International Conference on Nuclear Material Protection, Control and Accounting
Central Institute of Advanced Education, Obninsk, Russia
Hosts:
Federal Atomic Energy Agency of the Russian Federation State Scientific Centre of the Russian Federation—Institute for Physics and Power Engineering
Co-Sponsors:
U.S. Department of Energy National Nuclear Security Administration and the Institute for Nuclear Materials Management
Contact:
Leslie G. Fishbone
E-mail: fishbone@bnl.gov

June 5 – 9, 2005
ANS Annual Meeting
Town & Country Hotel and Resort
San Diego, California, U.S.A.
Sponsor:
American Nuclear Society
Contact:
American Nuclear Society
Web site: http://www.ans.org

July 7 – 8, 2005
Workshop on Developing Physical Protection Specialists
JW Marriott Desert Ridge
Phoenix, Arizona, U.S.A.
Sponsor:
INMM Physical Protection Technical Division
Contact:
David Lambert
865/574-3900
E-mail: lambertld@ornl.gov

July 10 – 14, 2005
46th INMM Annual Meeting
JW Marriott Desert Ridge
Phoenix, Arizona, U.S.A.
Contact:
INMM
847-480-9573
Fax: 847-480-9282
E-mail: inmm@inmm.org
Web site: http://www.inmm.org

August 7–11, 2005
ANS Topical on Decommissioning, Decontamination, and Reutilization
Hyatt Regency Hotel
Denver, Colorado, U.S.A.
Sponsor:
American Nuclear Society
Web site: http://ddrd.ans.org/index_meetings.html

October 30 – November 2, 2005
Changing the Safeguards Culture
Hotel LaFonda
Santa Fe, New Mexico, U.S.A.
Sponsor:
INMM International Safeguards Technical Division, INMM Southwest Chapter and ESARDA
Contact:
Jim Larrimore
Phone: 858/509-9604
E-mail: larrimor1@cs.com

December 11–14, 2005
European Nuclear Conference (ENC) 2005
Palais des Congres
Versailles, France
Sponsor:
French Nuclear Society (SFEN)
Contact:
Sylvie Delapace
Phone: +33 (0) 1 53 58 32 16
E-mail: enc2005@sfen.fr
Web site: http://www.sfen.fr