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# INTERNATIONAL MANAGEMENT OF SPENT FUEL STORAGE: TECHNICAL ALTERNATIVES AND CONSTRAINTS

Topical Report

by

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## ABSTRACT

Some of the important technical issues involved in the implementation of a spent fuel storage regime under international auspices are discussed. In particular, we consider: the state of the art as far as the different possible storage modes are concerned, the relevant accident, sabotage, and transportation considerations, and the impact of recent technical spent fuel safeguards initiatives on the non-proliferation rationale for international spent fuel management.

## I. Introduction

International management of spent nuclear reactor fuel has been proposed as a response to the following nonproliferation concerns:

1. Energy resource and waste management considerations have been advanced in support of the argument for closing the fuel cycle via reprocessing of spent fuel. A spent fuel storage regime is seen as providing a viable alternative to immediate reprocessing, while retaining this option for a future time when the cost/benefit tradeoffs involved in insuring a long-term fission option via the fast breeder reactor becomes clearer.

2. Large amounts of spent fuel in national hands, especially fuel which has been out of core for a long period of time, is seen as an invitation to nuclear mischief via covert or overt seizure followed by reprocessing in a dedicated facility.

In this report we briefly address some of the important technical issues involved in the implementation of a long-term, retrievable spent fuel storage regime. These issues can be summarized as follows:

1. In general, what is the state of the art as far as the different interim storage modes are concerned? In particular,

a. What is the nature of the tension between retaining the option for eventual reprocessing and the option for permanent disposal?

b. What length of time must spent fuel remain underwater before dry surface, near-surface or geologic storage becomes feasible?

2. What are the relevant accident, sabotage, and transportation considerations? How do these differentiate between the storage modes?

3. What impact would recent technical initiatives to upgrade IAEA safeguards on spent fuel stored in reactor water basins have on the nonproliferation rationale for international management?

4. How high is the spent fuel radiation barrier to diversion and reprocessing as a function of time?

These issues are addressed in the following. In particular, Section II is devoted to a technical description of the various storage concepts, while Section III and IV discuss environmental impacts and transportation. Our tentative conclusions and recommendations are summarized in Section V, while the safeguards and radiation barrier issues are taken up in Appendices A and B, respectively.

## II. Alternative Spent Fuel Storage Modes

For orientation we present in Figure 1 a generic view of the back end of the nuclear fuel cycle, which illustrates the role of the various interim storage modes. As indicated, the basic uncertainty at this time is associated with the role of the geologic mode. At first glance, the idea of engineering this mode, which is normally associated with ultimate disposal of high level waste, in a manner compatible with long term retrievability seems very attractive. (Short term retrievability, for safety purposes, is always a design requirement.) However, there are important technical considerations which argue for reserving the geologic mode for disposal, not storage. We consider this question, and the other uncertainties indicated in Figure 1, i.e., the length of time under water before dry storage becomes feasible and the packaging requirements associated with the different branches of the option space, in some detail below. Before beginning our discussion of the different storage modes, it is appropriate to note that the new element in the concept of interim storage at the back end of the fuel cycle is the emphasis on spent fuel rather than the solidified high level radioactive waste. Most of the storage modes discussed here for the former application have already been considered for the latter,[1,2] and much of the analysis is applicable with some modification.

## A. Storage Under Water

The handling and storage of radioactive materials under water is a standard method of operation in the nuclear industry. In particular, spent fuel has been routinely stored in water basins for many years; e.g., the first reactor pool--associated with operations of the Manhattan District--was put into service in the US in 1943, and the first commercial pools for storage of PWR, BWR and HWR spent fuel were completed in 1957, 1960, and 1962, respectively. While the concept of extended water storage of spent fuel is new, de facto, with the delay in reprocessing, both spent fuel and experience in storing it have been accumulating over the years, both in the US and abroad. In addition, extended water storage of solidified high level waste in canisters was considered by the National Academy of Sciences Panel on Engineered Storage and the AEC in 1974-75.[1,2] However, bare spent fuel differs from canned high level waste with respect to such factors as potential corrosion mechanisms and criticality, and specific questions have been raised regarding the integrity of spent fuel in extended water storage. An assessment of these concerns has been undertaken by A.B. Johnson, Jr. His reported results[3] and some related considerations are summarized in b below; to supply some context, we first review briefly the current practice in spent fuel water storage.

(a) The typical spent fuel water basin is a rectangular "tub" with walls below ground level composed of several feet of reinforced concrete lined with a water tight barrier such as stainless steel or fiberglass. Spent LWR assemblies are housed upright in racks mounted on the bottom of the pool; appropriate rack spacing and construction provide assurance against accidental criticality.

(Criticality precautions are unnecessary for spent HWR fuel stored under ordinary (not heavy) water, and the short (0.5m) fuel bundles are stacked in baskets with typically 32 bundles per basket.) The use of stainless steel or boron-impregnated stainless steel instead of aluminum racks permits roughly a factor of 1.5-2 decrease in rack spacing, and hence a factor of 2-4 increase in storage density. For example, the planned "reracking" of the U.S. Trojan PWR reactor pool with an original design capacity of 280 assemblies will permit the storage of approximately 650 assemblies; this corresponds to an increase in capacity from 4/3 to 10/3 of a full core (the normal annual discharge is 1/3 of a full core). The use of high density racks is an obvious, straightforward but limited solution to the growing shortage of space in reactor pools.

All spent fuel handling operations; e.g., transfer from the reactor to the racks in the reactor pool and any subsequent transfer to truck or rail casks for shipment to an Independent Spent Fuel Storage Installation (ISFSI), takes place under water. Cask loading in a reactor basin and loading and unloading in a water ISFSI takes place in a separate pool adjacent to the storage area which is designed to withstand accidental drop of a massive cask. Standard operating procedure also includes decontamination of incoming and outgoing casks before and after they are placed in the cask loading/unloading pool.

Adequate water pumping and heat exchanger capacity in a closed circuit system is provided to insure that the radioactive heat generated by the fuel--which can be quite high for fuel recently out of core -- does not cause the bulk water temperature to rise above $\sim 40^{o}C$. Radioactivity in storage pool waters is due to the presence of fission products from the fuel and neutron activation products from crud deposited on the fuel during reactor operation. Filtration and ion exchange are the principal methods for control of these species as well as other particulate and dissolved impurities such as chloride ion. The water pool chemistry is usually sampled on a weekly basis, and monitors for airborne radioactivity operate continuously above the pools. The radiation dose rate at the water surface of current generation pools is less than 1 mrem/hr.

Regulatory guidance specifying requirements for design, site selection (including acceptable geology, meteorology, hydrology, and water supply), and physical protection of pool storage facilities is based, in the U.S., on USNRC Guides 3.24 and 1.13 for independent pools and pools at reactors, respectively.

(b) The intuition that storage under water is a viable option for the long-term is usually inferred from the fact that the carefully controlled pool envrionment is much more benign than that existing inside a reactor core where the fuel has been sitting for periods in the range of one to three years (the shorter/longer times are characteristic of HWR/LWR fuel). Hence fuel that survived reactor exposure without developing defects would be expected to age gracefully under water, while defective fuel which fails during storage would do so in a noncatastrophic manner and could be isolated from the pool water in closed canisters. Johnson's survey[3] of U.S., Canadian, and European water storage experience and possible fuel and fuel cladding degradation mechanisms largely confirms this argument. In particular, he observes that:

(1) Both Zircalloy-clad and stainless steel clad uranium oxide fuel have been stored under water for long periods of time with no evidence that fuel bundle materials are degrading, based on visual inspections and radiation monitoring.[4]

Observed fuel failure rates are low (∿0.01-0.1%), and fuel assemblies with defective rods can usually be stored without special procedures. However, special equipment has been developed to handle failed fuel; e.g., containment of "leakers" in closed cannisters.

(2) Extrapolation of available experimental evidence suggests that known corrosion mechanisms such as cladding oxidation pose no threat to fuel integrity in water storage.

(3) Oxidation of $UO_2$ to $U_3O_8$ at fuel defects occurs very slowly at pool temperatures. However, the reaction rate increases rapidly with temperature, and at temperatures which may be attained in dry storage (≃300°C), substantial oxidation can occur, assuming, of course, that an oxidant is available. The relevance of this to various dry storage options is discussed below.

(4) To insure the credibility of water storage over the long-term, additional research is needed in several areas, including: possible effects of pool temperature and water chemistry transients--such as might occur during loss-of-

cooling capability accident--on the subsequent condition of stored fuel, the behavior of fuel defects as a function of defect type, cladding type and storage conditions, the effects of galvanic couples on the hydriding of zirconium alloys, and the definition of special effects such as crud layer environments and crevice corrosion. A low-level, selective spent fuel surveillance regime should be instituted as an integral part of this research program.

In conclusion, it appears that optimism regarding extended water storage is justified. The technology is well-established, and this mode is the logical reference case since water storage is the inevitable first step in any spent fuel regime. To be sure, no absolute assurance that spent fuel can be stored retrievably either under water or in a passive mode for periods up to 100 years can be given today, and in this sense statements--made; e.g., at the recent Windscale inquiry in support of the application of British Nuclear Fuels, Ltd. (BNFL) to build on oxide reprocessing plant--to the effect that long term storage <u>may</u> lead to severe and costly deterioration problems, cannot be dismissed out of hand.

However, at least for the mid-term (up to say 25 years), the weight of available scientific evidence and operational experience in water storage inspires confidence that immediate reprocessing is not a <u>technical</u> requirement to insure ultimate retrievability of spent fuel.

## B. Dry Storage

Dry storage of spent fuel becomes a viable option after it has cooled to the point where passive heat transfer from encapsulated fuel to its environment is efficient enough to insure that the fuel element temperatures are well below values which would lead to significant degradation over the long-term. Most of the known failure mechanisms are strongly temperature dependent. For example, as previously noted, recent experimental evidence[5] indicates that $UO_2$ exposed at a cladding defect to temperatures ≈300°C will rapidly oxidize in air to $U_3O_8$, causing swelling and splitting of the cladding with exposure of more fuel, and ultimately release of finely powdered $U_3O_8$ and fission products. To avoid this situation, one can either restrict the out-of-core age of fuel/encapsulation method "space" to insure that fuel temperatures remain well below 300°C, or eliminate the risk of fuel oxidation by replacing the air atmosphere inside the canister containing the fuel with an inert environment. Current thinking--based on moving fuel from reactor basin to dry storage as soon as possible, and simplifying the procedure for eventual reprocessing--favors the latter course with helium filled canisters (helium also provides convenient leak detection). However, a view of dry storage as a *de facto* disposal mode would dictate different design requirements. This example illustrates the issues involved in engineering what is basically--as compared to water storage--a new, albeit straightforward technology.

Three basic techniques for surface or near-surface storage are presently being considered in the U.S., Canada, and Western Europe: the dry well or caisson, the sealed cask, and the air-cooled vault. In the following we briefly discuss these modes as well as the concept of retrievable geologic storage.

The Dry Well or Caisson (Fig. 2)

In the dry well or caisson concept, canned fuel is stored below ground level in lined, vertical shafts which are sealed at the top for radiation shielding purposes. The rationale for this approach is that reliance on the soil for heat dispersion by conduction to the ground surface, for radiation attenuation, and for physical protection, minimizes capital and operating costs, and provides a safe, economic alternative to long-term under water storage with easy retrievability. On paper, this concept--which has been developed over the past several years by the Atlantic Richfield Company and is now also being studied by the U.S. D.O.E.[6]--looks quite attractive. The storage holes would be constructed as needed on a square grid; with a spacing of 20-25 feet, storage of the spent fuel from $\sim$1 GWe-yr of LWR operation would require approximately one acre of land, assuming one PWR assembly per dry well. Heat transfer is by radiation and convection (primarily the former for young fuel) from the fuel to the can and from the can to the hole liner, and then by conduction through the soil to the atmosphere. The major design variables which determine the permissible decay heat load--and hence the age and amount of fuel which can be stored--are the soil conductivity, the hole spacing, and the hole diameter. Fig. 3, taken from Reference 6, illustrates the strong dependence of the can temperature on the soil conductivity. It has been claimed[7] that spent fuel aged 2-3 years under water can be stored in wells spaced 25 feet apart without exceeding a fuel cladding temperature of $\sim 380°C \simeq 715°F$, assuming a soil conductivity of 0.2 Btu/hr-ft-°F. (The indicated, allowable temperature is based on not exceeding two-thirds of the cladding rupture stress of PWR fuel rods.)[6] However, this result is derived from computer heat transfer analysis and awaits experimental confirmation. The heavy reliance on the soil is both the basic attraction and a potential drawback of this storage concept.

<u>The Sealed Cask (Fig. 4)</u>

In contrast to the dry well, in the storage cask method fuel is stored in units which are self-shielding, cooling and protecting. Each cask is a hollow reinforced concrete cylinder sitting on a concrete base. The spent fuel is contained in a steel can at the center of the cylinder, with possibly a thin lead layer between the fuel can and the concrete walls. (Lead is a much better shielding material than concrete, and its use reduces the thickness of concrete required.) In the U.S., a similar design was recommended by the National Academy Panel on Engineered Storage[2] as the optimum method for interim storage of high level radioactive waste (the dry well was not considered). In this application, natural draft air circulation through an annulus between the shield and the fuel can was necessary to remove the heat being generated at the design rate of 5 KW. However, since the heat rate per unit volume of aged spent fuel is lower than that of high level waste,[8] a sealed cask design seems feasible for the former. This route has been actively pursued at the Whiteshell Nuclear Research Establishment (WNRE) in Canada, where a development and demonstration program was started in 1974.[9] Tests at WNRE have verified that at a design heat load of 2KW, corresponding to storage of 4.4MT of CANDU HWR fuel cooled for five years, the fuel cladding temperature is acceptably low (≈200°C). (A similar heat load would be obtained from ∿1.6MT of five-year cooled LWR fuel.) Besides fuel integrity, the other major potential materials problem associated with the cask concept is degradation of the concrete shield due to thermal stresses. That is, at heat rates in the range 1.5 to 2KW, the high temperatures at the internal concrete wall result in tensile stresses on the outside surface which exceed the maximum tensile stress of the concrete. Under these conditions, surface cracking is predicted; however, it was expected that the reinforcing steel would prevent the cracks from growing to the point where the structural or shielding integrity of the concrete would be compromised. Initial tests[9] have verified that surface cracking is minimal, even at higher than design heat loads (≈5KW) with the casks subjected to simulated freeze-thaw cycles. (The fact that heat transfer through the concrete was much better than predicted undoubtedly was a major factor in these results.)

These results are encouraging, but more work is needed to insure the ability of the concrete to provide an effective shield for periods ≈50 years. In this regard, treating the outer surface to increase its emissivity (thus lowering the heat load from the sun and increasing the permissible decay heat load) and to slow down weathering would be a major improvement.

## The Air-Cooled Vault (Fig. 5 )

In comparison to water cooling, both the dry well and the sealed cask are low density storage modes. (The minimum spacing of sealed casks--on the order of 25 feet--is determined more by the need for access than heat transfer considerations.) A dry storage concept with a packing density of the same order as water storage is the air-cooled vault. In this system, the fuel is contained within canisters and secondary canisters, or overpacks, and stored in closely spaced vertical stacks in a large concrete bunker which is constructed partially below grade to reduce the radiation shielding and physical protection requirements. Heat removal occurs by natural convection of air flowing directly through the vault with the chimney effect of the hot air rising from the fuel providing the circulation. As with water storage and the concrete cask, the vault was previously considered in the U.S.[1,2] for the storage of high level waste, and is now being reconsidered along with the dry well and concrete cask by the U.S. D.O.E. in their Spent Unprocessed Fuel (SURF) Facility Program. The convection vault concept has also been studied in Canada along with a variation, the conduction vault.[10] In both schemes, the CANDU fuel bundles are precast into zinc cylinders in an aluminum mold before canning to improve the containment and heat transfer. In the conduction vault variation, the fuel canisters are stacked tightly together and closed at the top by a finned aluminum shield plug. Heat flows up the canisters by conduction through the zinc castings and is dissipated to the air by the shield plug fins. No cooling air enters the fuel area of the vault. This reduces the possibility that activity could become suspended in the exhaust air, at the "price" of less efficient cooling as compared with the convection vault. Important uncertainties with both schemes include: can the zinc be melted easily to allow recovery of the fuel, how much reduction in air flow can be tolerated, and is there any interaction between the zinc and zircaloy during casting or later in storage?

## Geologic Retrievable Storage

Although the tension between reprocessing and ultimate disposal arises in all dry storage modes, it is felt most keenly in the case of geologic storage. On the one hand, it has long been appreciated that the plasticity and good thermal conductivity of salt make salt beds an attractive candidate medium for ultimate disposal of radioactive waste. On the other hand, the corrosive attack of included brine which tends to migrate up thermal gradients towards the heat source, and mechanical deformation of the salt at high temperatures demands a high degree of conservatism in packaging and thermal loading in order to keep open the retrievability option for an extended period of time. For example, DOE has recently estimated[11] that the thermal loading of a generic salt repository at a depth of 2000 feet must be restricted to 36KW/acre if 25-year retrievability is to be assured, while keeping the retrievability option open only for a five year initial "shakedown" period would allow densities of ]50KW/acre. For non-salt formations such as basalt, granite, and shale, corrosive attack is not known to be a problem (current knowledge of canister-rock interactions is meager), but similar heat loading retrievability restrictions exist. Although it would be nice "to have your spent fuel and dispose of it too," the geologic mode is probably best reserved for ultimate disposal of either spent fuel or high level waste.

## III. Environmental Impact

The greatest potential hazard to the public from stored spent fuel is the release of radioactivity due to either missile impact or containment failure via loss of coolant. However, even if the radioactive release in some accident, natural disaster, or sabotage attempt were limited, the retrievability option might still be significantly degraded and a large amount of fuel might have to be relocated in a short period of time. Given the relative state-of-the-art in water and dry storage, one would expect that statements relating to the environmental impact of the former would be much more definitive. This situation is reflected in the draft generic environmental statement recently issued by the US Nuclear Regulatory Commission.[12] Indeed, the discussion of normal and abnormal events and their consequences is entirely confined to water storage. Their basic conclusions can be summarized as follows:

(1) Pool water and air quality can be easily monitored (there is essentially only one process stream), and the environmental impact of normal operation is nil (e.g., the volume of wastes associated with water cleanup is about $2m^3$/GWe-yr with a maximum associated radioactivity of about 10 Ci/GWe-yr of beta-gamma activity).

(2) The environmental impact of such events as: fires, explosions, earthquakes, missile accidents (e.g., tornado-driven utility poles penetrating the storage building and landing in the pool), and accidental criticality is small. This confidence is based primarily on the fact that the fuel sits below grade, under a minimum of 12 feet of water, surrounded by walls of reinforced concrete which are typically six feet thick. Moreover, besides the basin itself all important auxiliary equipment such as fuel handling cranes and crane supports are designed to prevent collapse of structures into the pool which could damage the fuel.

(3) A loss of cooling capability is potentially serious, but the large heat capacity of the pool water should provide adequate time for corrective action to be taken. We illustrate this point with an idealized (calometric) calculation:

Assume that a 30 foot deep, 1,000 MT capacity storage pool contains $10^6$ gallons of water and is loaded to capacity with fuel whose average heat rate is 10W/kg, corresponding to PWR fuel which has been out of core for one year. Then the total heat rate is

$$1000 \text{ MT} \times \frac{10\text{kw}}{\text{MT}} = 10^7 \ \frac{\text{joules}}{\text{sec}} \times \frac{3.6 \times 10^3 \ \text{sec}}{\text{hr}}$$

$$= 3.6 \times 10^{10} \ \frac{\text{joules}}{\text{hr}} = 3.4\text{x}10^7 \ \text{Btu/hr},$$

and the resultant temperature rise

$$\frac{3.6 \times 10^{10} \ \frac{\text{joules}}{\text{hr}}}{\frac{4.18 \ \text{joules}}{\text{gm} \ ^\circ\text{C}} \times \frac{3.77\text{x}10^3\text{gm}}{\text{gallon}} \times 10^6 \ \text{gallons}} = 2.3^\circ\text{C/hr}.$$

Assuming the ambient water temperature was 40°C at the time cooling was lost, it would take roughly 60°C/2.3°C/hr = 26 hours to reach boiling. To maintain the water level under boiling conditions would require makeup water to be supplied from an emergency source at a rate of

$$\frac{3.6\times10^{10}\ \frac{\text{joules}}{\text{hr}} \times \frac{1\ \text{hr}}{60\ \text{min}}}{\frac{540\ \text{calories}}{\text{gm}} \times \frac{4.19\ \text{joules}}{\text{calorie}} \times \frac{3.77 \times 10^{3}\ \text{gm}}{\text{gallon}}} \simeq \frac{70\ \text{gallons}}{\text{min}}$$

This is a modest requirement, and could be supplied from several sources; e.g., the ultimate heat sink. However, if makeup was not supplied, the top of the fuel assemblies would begin to be exposed in about 4 days.

To these remarks, we would add the following preliminary observations:

(1) Almost by definition, a dispersed storage mode would be less affected by abnormal events of limited geographical scope, and this would tend to favor the dry well and sealed cask as compared with pool storage and air-cooled vaults. In particular, low seismicity would be a site selection criterion for all modes, but it would be of greater importance for the latter. The "other side of the dispersion coin" is that the task of surveillance and monitoring for evidence of containment deterioration and abnormal activity becomes more difficult.

(2) As a corollary to the above, canned spent fuel either in a hole in the ground, or surrounded by a thick concrete shield would be relatively immune to severe damage via tornado, earthquake, airplane crashes, or sabotage using conventional explosives. The cask might be toppled off its pedestal and/or cracked, but the enviromental impact would be small.

(3) Besides the fuel cladding all dry storage modes rely on multiple barriers to contain possible radioactive releases. For all modes, more materials research and development is needed to assure the long-term reliability of these barriers, taking into account the possibility of accelerated rates of corrosion of metals or weathering of concrete under abnormal conditions.

(4) Of the dry storage modes, both loss of cooling capability, and criticality accidents would be more credible, and potentially more serious in the case of the convection vault.

## IV. TRANSPORTATION

An LWR shuts down annually to replace about 1/5 to 1/3 of its core. Storage of the discharged fuel at the reactor pool for a period of 150 days permits radiation levels and decay heat loads to diminish by roughly a factor of 100 as compared to levels immediately after discharge, and makes feasible shipment in massive casks (called "flasks" in England) which incorporate gamma and neutron shielding and provide cooling for the fuel. As with pool storage itself, shipping spent fuel in casks is not a new technology; i.e., present cask designs have evolved from experience gained since the mid-1940's in shipping fuel from commercial, military, and research reactors. There has been comparable experience in Europe, where from 1966 through 1975 about 590 MT of spent LWR fuel was shipped to reprocessing facilities.[13] Casks are usually classified according to the primary transport mode (truck or rail; a few designs can go either way) and the nature of the coolant (water or air). Truck/rail casks now available or under construction weigh up to 35 MT/100 MT fully loaded. Naturally, the large rail casks have a higher fuel capacity and their use substantially decreases the number of shipments and the loading and unloading capacity required for a given amount of fuel. For example, the Nuclear Fuel Services NFS-4 truck cask weighs 22 MT, has a fuel capacity of 0.5 MT U (uranium), (1 PWR/2 BWR assemblies), and a 2% payload while the corresponding figures for the National Lead Industries NLI

10/24 rail cask are 97 MT, 4.7 MT U (10 PWR/24 BWR assemblies) and 5%, respectively.[12] Unfortunately, not all reactors have access to rail facilities--only about half of those in the US--and truck transport is now and will continue to be an inevitable part of the spent fuel transportation picture. An aspect of the use of small capacity casks which is relevant to the rate of removal of spent fuel from basin storage is the typical cask turn-around time for shipping to international storage facility. Assuming that two NSF-4 type casks are available, and that it takes:

(a) approximately one day for loading, decontamination, sealing, checking, etc. of each cask before shipment from a PWR reactor pool,

(b) about the same amount of time for similar operations at the other end, and

(c) about four days in transit,

it would take more than a year from the time of reactor discharge for all 64 PWR spent fuel assemblies to have been shipped off the reactor site, assuming shipment to have started 150 days after discharge. The purpose of this example is to illustrate the potential magnitude of the "dead-time" transport problem. Not only does the fuel need someplace to go, but it takes some time to get there.

Related to the above is the question of cask availability. With the delay in reprocessing, the economic incentive for building casks has declined, and the number presently available/under construction--13/9 in the US and 14/6 in Western Europe--is insufficient for shipment of large quantities of spent fuel. However, there is no reason to believe that--given sufficient priority and economic incentive--fabrication of casks according to designs already licensed, should be a major bottleneck. (The NRC has estimated fabrication times of 10 months to 3 years for a truck cask, and from 1.5-4 years for a rail cask.[12]) In order to receive a license for a new cask, an applicant must demonstrate to the satisfaction of the Nuclear Regulatory Commission (NRC) and the Department of Transportation (DOT) that the cask provides required containment, shielding, criticality control, and heat transfer under both normal and accident conditions. In particular, a detailed Safety Analysis Report (SAR) must be filed with the NRC to demonstrate compliance with the applicable code, 10 CFR Part 71. (Similar IAEA requirements are detailed in the "Regulations of Radioactive Materials, Safety Series No. 6.") Since transportation accidents usually involve some combination of impact, puncture, fire, or submersion in water, the acceptance tests require evaluation of the cask and its contents for a 30-foot drop onto a completely unyielding surface, followed by a 40-inch drop onto a 6-inch diameter pin, followed by 30 minutes exposure to 1,475°F, followed by 24 hours of immersion in water. This is a formidable challenge; however, more extreme scenarios, some involving malevolent acts, can be imagined, and a breach of the cask containment with release of radioactivity near a highly populated area could have serious consequences. Per vehicle mile estimates of the probability of accidents of varying severity and their consequences in terms of population radiation dose are derived in WASH-1238, "Environmental Survey of Transportation of Radioactive Materials To and From Nuclear Power Plants," and are summarized in reference 12. As might be expected, the accident probability vs. consequence curves follow the pattern familiar from the Reactor Safety Study, WASH-1400; i.e., serious accidents occur via a series of improbable events, and hence have a very low overall probability. This is reassuring; however, location of any international facility to minimize the total shipping required and especially that near populated areas makes good sense.

## V. OBSERVATIONS

Consideration of extended spent full storage has become necessary with the delay in reprocessing,while the latter has been driven, especially in the U.S., by concern about weapons proliferation via separated plutonium. Thus it seems fitting to conclude this brief overview with some remarks which focus on the non-proliferation implications of the technical aspects of spent fuel management.

1. Spent fuel which has been out of core for less than ∿100 years is still protected by a radiation barrier which necessitates remote handling, and hence is not as vulnerable as stockpiles of separated, decontaminated plutonium. (The radiation barrier decreases sharply after this time due to the decay of Cs-137, the principal gamma emitter, which has a 30 year half-life). However, the decrease in the level of the radiation barrier in time (see Appendix B, Table 2) might make diversion of spent fuel which is more than ∿5 years old somewhat easier.

2. Cooling spent fuel at the reactor basin for at least one year makes good sense in terms of utilization of at-reactor storage capacity and in view of the problems associated with shipping intensely radioactive materials. Waiting for approximately five years would be even better for ease of transport (smaller, cheaper casks) and also makes available the option of moving directly from national wet storage to international/multinational passive dry storage with a high degree of confidence in the long-term integrity of the stored fuel. Moving from wet to dry storage at an earlier time is probably technically feasible, but would require reoptimization to enhance heat transfer, and this would increase the cost and also might complicate retrievability if additional packaging is required.

3. The ability of at-reactor pool spent fuel surveillance to provide timely, unambiguous verification of attempted diversion can and should be significantly enhanced irrespective of the fate of initiatives for international storage. It is often argued that the political hurdles involved in instituting tougher safeguards are formidable; perhaps, but similar problems are involved in establishing an international/multinational spent fuel storage regime or bilateral spent fuel return arrangements; e.g., the singular lack of enthusiasm of governments for welcoming fuel irradiated in foreign reactors. The point is that some spent fuel will always be in national hands, and non-intrusive electronic surveillance techniques, while not foolproof, can make a major contribution to nonproliferation.

4. Despite its attractive features, i.e., a well-developed technology which permits high storage concentration with relative ease of access, water storage has the disadvantage of requiring active cooling and cleanup. This leads to higher costs for perpetual care compared with dry storage modes, and greater vulnerability in the event that all supervision is lost for an extended period of time because of unstable political conditions, natural catastrophies, etc. Hence, if water basins are chosen as the centralized storage mode, consideration should be given to locating them underground in order to at least partially offset these vulnerabilities. This should not involve great additional expense, since, as previously noted, current practice is to build pools partially below ground.

5. Both the dry well and the concrete cask concepts are attractive as backup to water basis for interim storage of spent full or radioactive waste. Both require more development and testing.

6. As previously indicated, transportation of spent fuel should not be a major problem from the point of view of cask requirements, environmental impacts, or cost. However, multiple shipments with attendant rehandling increase all these factors, and the public anxiety about releases of large amounts of activity - especially via terrorist attack - makes the optimization of interim storage logistics a priority item. The "obvious" solution is colocation of interim storage either with reprocessing facilities or geologic formations suitable for disposal, but which one? The attractive feature of the latter is the potential for ready conversion to the former without additional transport, if a decision is made to close the fuel cycle in this manner. However, tying these concepts together would preclude early implementation of an extended storage regime because of the exacting technical site selection requirements for disposal. On the other hand, the viability of an extended storage regime would be compromised by locating it "conveniently" near an existing or planned reprocessing facility; e.g., Cap La Hague or Windscale. These considerations. taken together with the political problem of finding suitable national sites for a multinational storage facility, have led to the suggestion(14) that a remote, sparsely populated island would be a desirable site--initially for storage, and perhaps later for other fuel cycle activities; e.g., reprocessing and production of methanol via fast reactors. The political and technical problems involved in finding suitable locations have led some to characterize this concept as a "pie in the ocean." However, it should not be dismissed out of hand, if only because of the paucity of possible alternatives. For an interesting discussion of the particular siting issues involved in the Indian Ocean area, see reference 15.

# APPENDIX A

## SPENT FUEL SAFEGUARDS

The criteria for required levels of physical protection of nuclear materials under the IAEA regime is spelled out in INFCIRC/225, Rev. 1, June 1977. There are three categories: I, II, III, in order of decreasing stringency of safeguards. In particular, both spent fuel and unirradiated natural or slightly enriched fresh fuel are in category III which provides for:

Use and Storage wtihin an area to which access is controlled.

Transportation under special precautions including prior arrangements among sender, recipient and carrier, and prior agreement between entities subject to the furisdiction and regulation of supplier and recipient states, respectively, in case of international transport specifying time, place and procedures for transferring transport responsibility. What this means in terms of current surveillance of spent fuel pools is the following: The condition of the pool is monitored by either movie or video cameras on a semi-continuous basis; i.e., once every 15-30 minutes, on the presumption that this period is short compared with the time required to move fuel. The video tapes are inspected every 3 months, and the movie film every 6 months. In addition, access to the pool area is monitored by gamma detectors. This is clearly not the last word in safeguards because: (1) it is possible to tamper with the transmission link between camera and tape recorder, and (2) the inspection interval does not provide timely warning. Various technical initiatives are under development which, if implemented, would significantly increase the timeliness of warning of attempts to divert fuel and/or tamper with the safeguard system. In particular, a prototype remote surveillance and interrogation system has been designed and is now being tested.[1] This system can be operated in various ways to monitor the status of a storage pool in almost "real-time;" e.g.,

(a) a TV camera takes a picture of the pool. The picture is stored as a reference in a memory device. Every minute or os, the camera takes another picture and compares it with the one stored in memory. If something has changed, a status module on the camera transmits an alarm to an on-site multiplexer and from there to a remote verification unit in; e.g., Vienna via telephone cable. The primary technical problem is a high false alarm rate caused by the difficulty in discriminating between benign events; e.g., changes in lighting over the pool and the alarm conditions. The same basic system could be used to send pictures of the pool in almost real time by recording the video, digitizing it, compressing the bandwidth and transmitting via telephone cable about a minute later. This technique is called slow scan video.

(b) A fiber optic cable can be entertwined through all the spent fuel assemblies and connected directly to a status module which senses continually in the cable, and transmits an alarm if there is a break to the on-site multiplexer and then on to Vienna as above.

The complete system consists of the monitoring units, on-site multiplexer, and remote verification unit, and was developed by Atlantic Research under contract to ACDA. It can accept inputs from a variety of sensors. The motion detector is built by Fairchild Camera.

Besides this work, special procedures have been developed by Atomic Energy of Canada, Ltd. (AECL) and the IAEA for safeguarding the on-power fuelling feature of the safeguard system are:

- fuel bundle counters which count the number of bundles being discharged from the reactor into the storage bay via the fuel transport system
- cameras within the reactor building which can detect the removal of bundles from the reactor if this is done in any way other than via the fuel transport system
- a bundle radioactivity monitor to verify that the bundles in the storage bay have been irradiated and are not dummies.

Further details can be found in reference 2.

# REFERENCES

1. For technical details, see draft report. "RECOVER - Remote Continual Verifications," Atlantic Research Corp., July 1977.

2. A. Waligura et al, "Safeguarding On-Power Fuelled Reactors - Instrumentation and Techniques, Paper IAEA - CN 36/185 presented at the IAEA International Conference on Nuclear Power and its Fuel Cycle, Salzburg, Austria, May 2-13, 1977.

# APPENDIX B

## THE SPENT FUEL RADIATION BARRIER

The attractive feature of spent fuel from the nonproliferation perspective is that access to the contained plutonium is inhibited by the intense gamma radiation field of the decaying fission products. We illustrate how this barrier decays as a function of time after discharge by considering the properties of the spent fuel from a large 1150 MWe Westinghouse PWR of current design. The relevant reactor characteristics are:[1]

1. Fuel Burnup: 33,000 MWD/MTU
2. Specific Power: 37.8 MW/MTU
3. Average Fresh Fuel Enrichment: 2.6% U-235
4. Square Fuel Assemblies
    a. Side Dimension: 21.4 cm
    b. Active Fuel Length: 366 cm
    c. Weight of Uranium: 520 kg.

Voluminous data on the nuclide concentrations and gamma decay energy from U-235 irradiated at a specified thermal neutron (2,200 m/sec) flux, $\phi$, for a specified time, $\tau$, and then allowed to decay is available;[2] to extract the appropriate numbers we use the given reactor characteristics to compute:

$$\frac{\text{U-235 atoms}}{\text{assembly}} = \frac{\frac{26\text{ kg}}{\text{MTU}} \times \frac{6 \times 10^{26}\text{ atoms}}{\text{kg mole}}}{235\text{ kg/kg mole}} \times \frac{0.52\text{ MTU}}{\text{assembly}}$$

$$= 3.46 \times 10^{25}$$

$$\phi = \frac{37.8 \text{ MW/MTU} \times 3.2 \times 10^{16} \text{ fissions/sec/MW}}{6.66 \times 10^{25} \text{ atoms/MTU} \times 580 \times 10^{-24} \ \frac{\text{cm}^2}{\text{atom}}}$$

$$= 3.13 \times 10^{13} \text{ cm}^{-2} \text{ sec}^{-1}$$

$$t = \frac{33{,}000 \text{ MWD/MTU}}{37.8 \text{ MW/MTU}} = 873 \text{ days} \cong 7.54 \times 10^{7} \text{ sec}$$

From Fig. T-11, 11a of Reference 2, we can now find the gamma power in watts/assembly, S, for some representative times, $\tau$, after discharge:

Table 1

| t | S |
|---|---|
| Discharge | $15.2 \times 10^4$ |
| 1 month | $17.3 \times 10^3$ |
| 150 days | $3.5 \times 10^3$ |
| 1 year | $12.1 \times 10^2$ |
| 5 years | 208 |
| 10 years | 138 |
| 30 years | 121 |

To convert S into gamma flux, I, we model the assembly as a line source of length 366 cm. Then I in watts/cm$^2$ one meter from the midplane of an assembly is given by

$$I = 2 \int_0^{183 \text{ cm}} \frac{S/366}{4\pi(x^2+100^2)} \, dx = \frac{S \tan^{-1} 1.83}{2\pi \times 366 \times 100}$$

$$= 4.7 \times 10^{-6} \text{ S/cm}^2.$$

To take into account self-absorption by the 264 fuel pins in the assembly, we average the mass of the fuel over the entire assembly volume and use the mass absorption coefficient characteristic of the 0.66 Mev fission product gammas from Cs-137, which makes the principal contribution to the gamma activity after 150 days. This gives an average reduction in I of approximately a factor of 5. Thus

$$I \rightarrow 0.94 \times 10^{-6}\ S/cm^2.$$

Finally, to get the gamma dose corresponding to the foregoing gamma flux we again assume that all the fission product gammas have the effective energy of those from Cs-137, which have a mass absorption coefficient in water of 0.032 $cm^2/gm$. Since a dose of one rad represents absorption of 100 ergs/gm, the close rate D in rad/hr from a gamma flux I in $watts/cm^2$ is

$$D(rads/hr) = \frac{I(watts/cm^2)\ 0.032(cm^2/gm)\ 10^7 (\frac{erg}{watt/sec}) x 3.6x10^3 (sec/hr)}{100\ (ergs/gm\ rad)}$$

$$= 11.5 \times 10^6\ I\ (watts/cm^2)$$

$$\simeq 10S\ (watts/assembly)$$

Hence, from Table 1 the gamma dose 1 meter from the midplane of our PWR assembly at time t after discharge is:

Table 2

| $\underline{t}$ | $\underline{D}$ (rads/hr at 1 meter) |
|---|---|
| Discharge | $1.5 \times 10^6$ |
| 1 month | $1.7 \times 10^5$ |
| 150 days | $3.5 \times 10^4$ |
| 1 year | $1.2 \times 10^4$ |
| 5 years | 2000 |
| 10 years | 1400 |
| 30 years | 1200 |

To get a feel for the barrier represented by these numbers, we note that complete incapacitation begins at $\sim 10^4 - 2 \times 10^4$ rem (equivalent to rad for gamma rays at these energies), while exposure to about 500 rem will result in one-half of the individuals so exposed dying. (This is the so-called LD 50 dose.) Below about 200 rem there are no discernible near-term effects. From the point of view of ease of commercial reprocessing via the Purex process, less shielding would be required for old fuel, and there would be less of a problem with radiation degradation of the organic solvent, tributyl phosphate, and with attaining commercial Pu decontamination levels since; e.g., the hard-to-separate fission products zirconium and niobium would have decayed to insignificant levels. The relevance of these matters to reprocessing in a dedicated facility is not clear.

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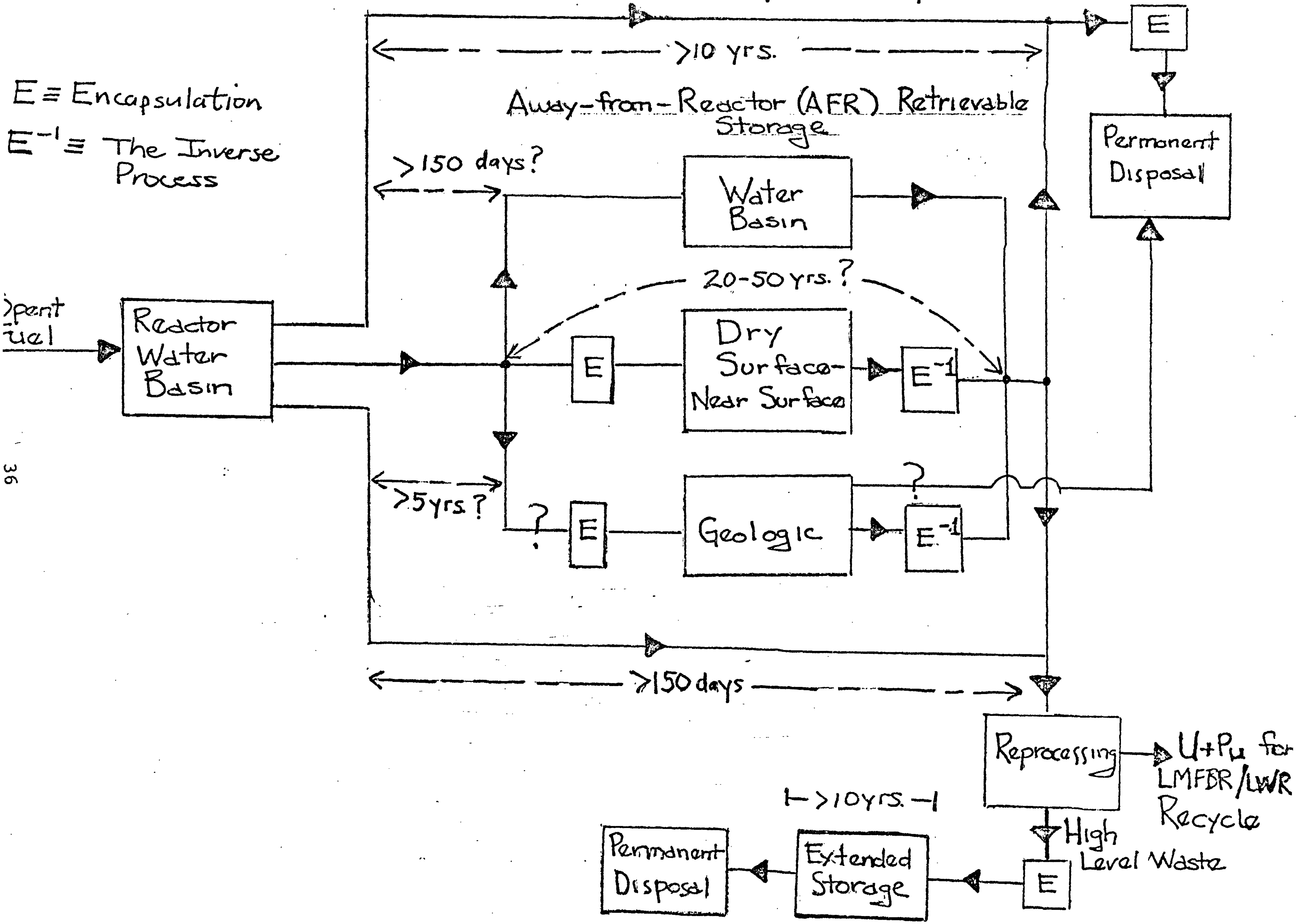
1. J.J. Duderstadt and L.J. Hamilton, Nuclear Reactor Analysis, John Wiley and Sons, 1976, p. 634.
2. J.O. Blomeke and M.F. Todd, Uranium-235 Fission-Product Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time, Oak Ridge National Laboratory Report ORNL-2127, August 1957.

# Figure 1. The Back-End Option Space



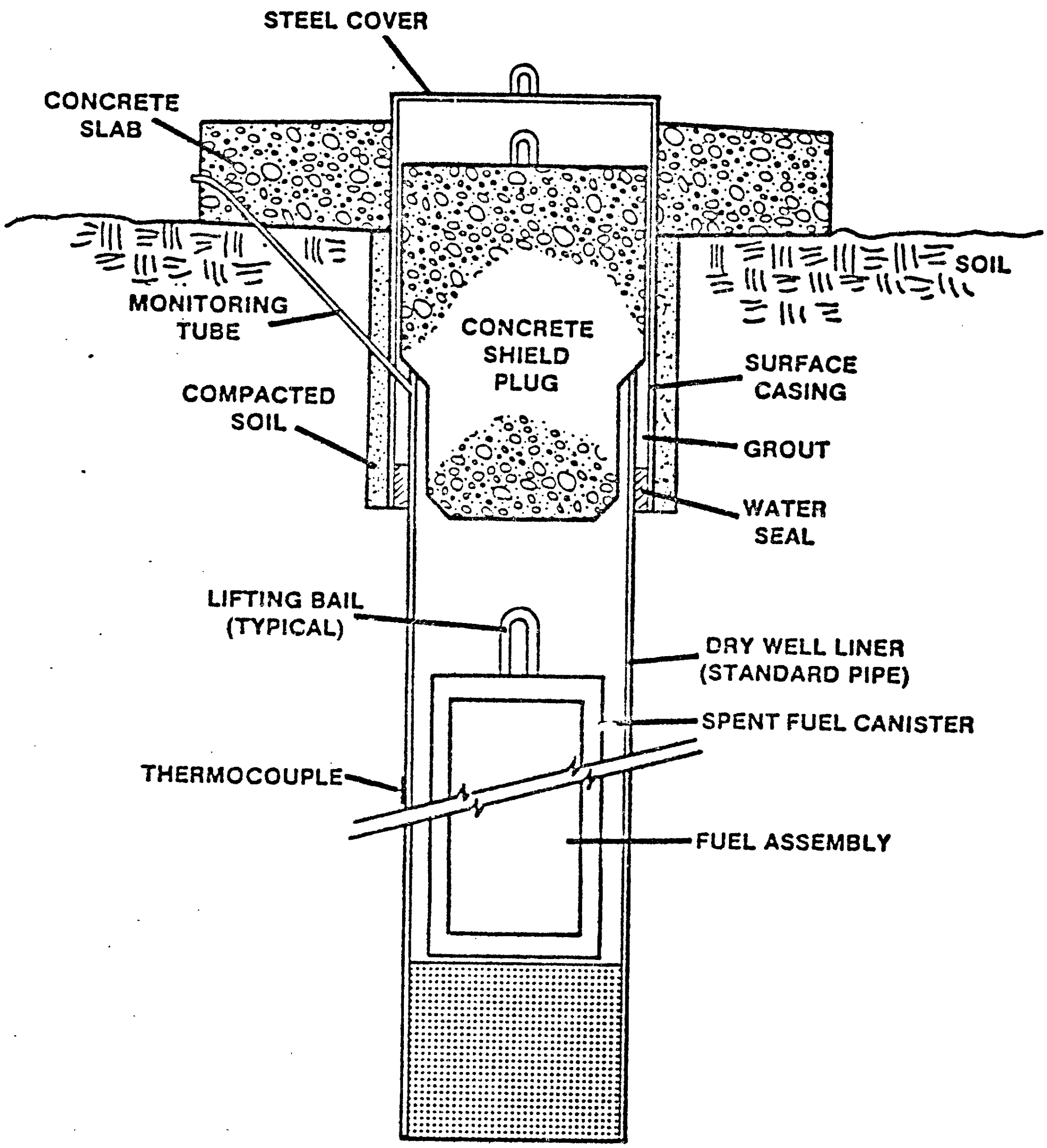



FIGURE 2: Dry Well Concept

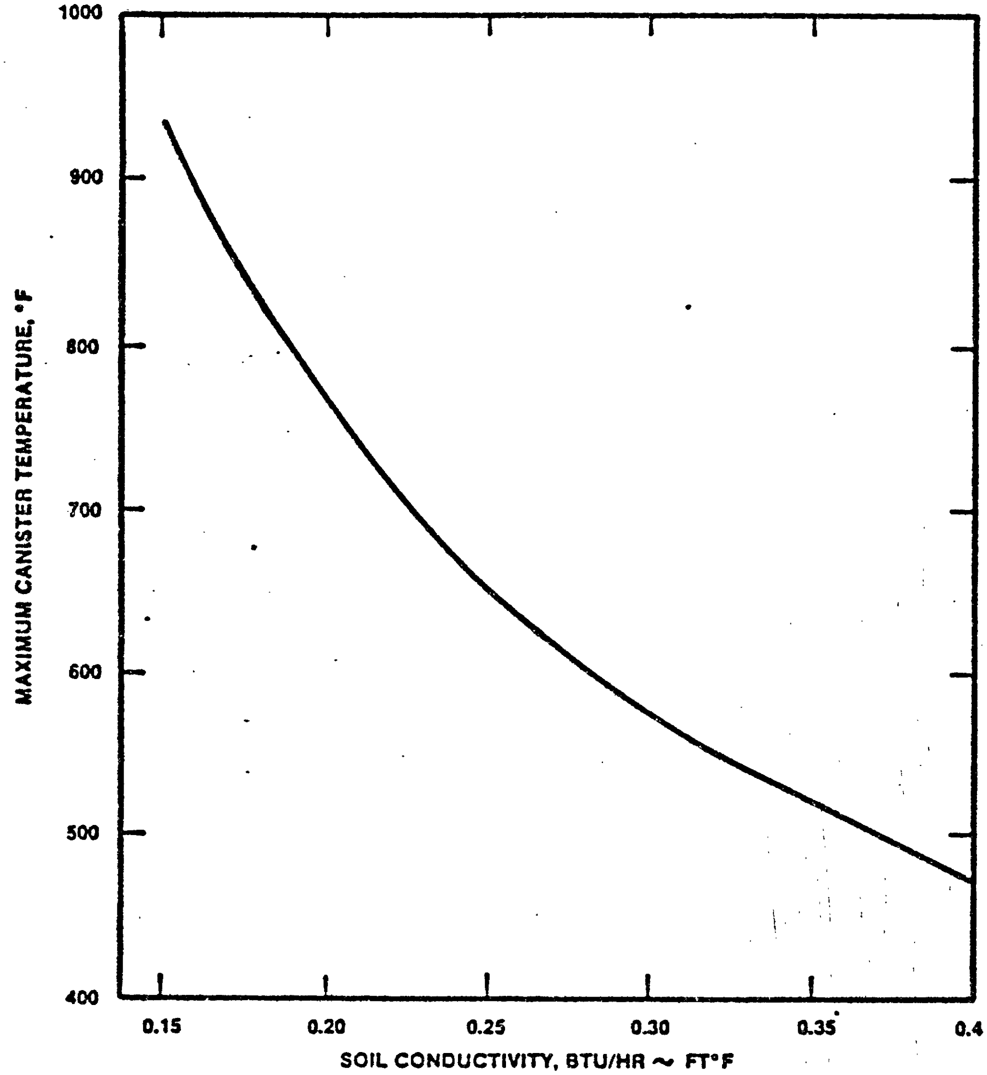
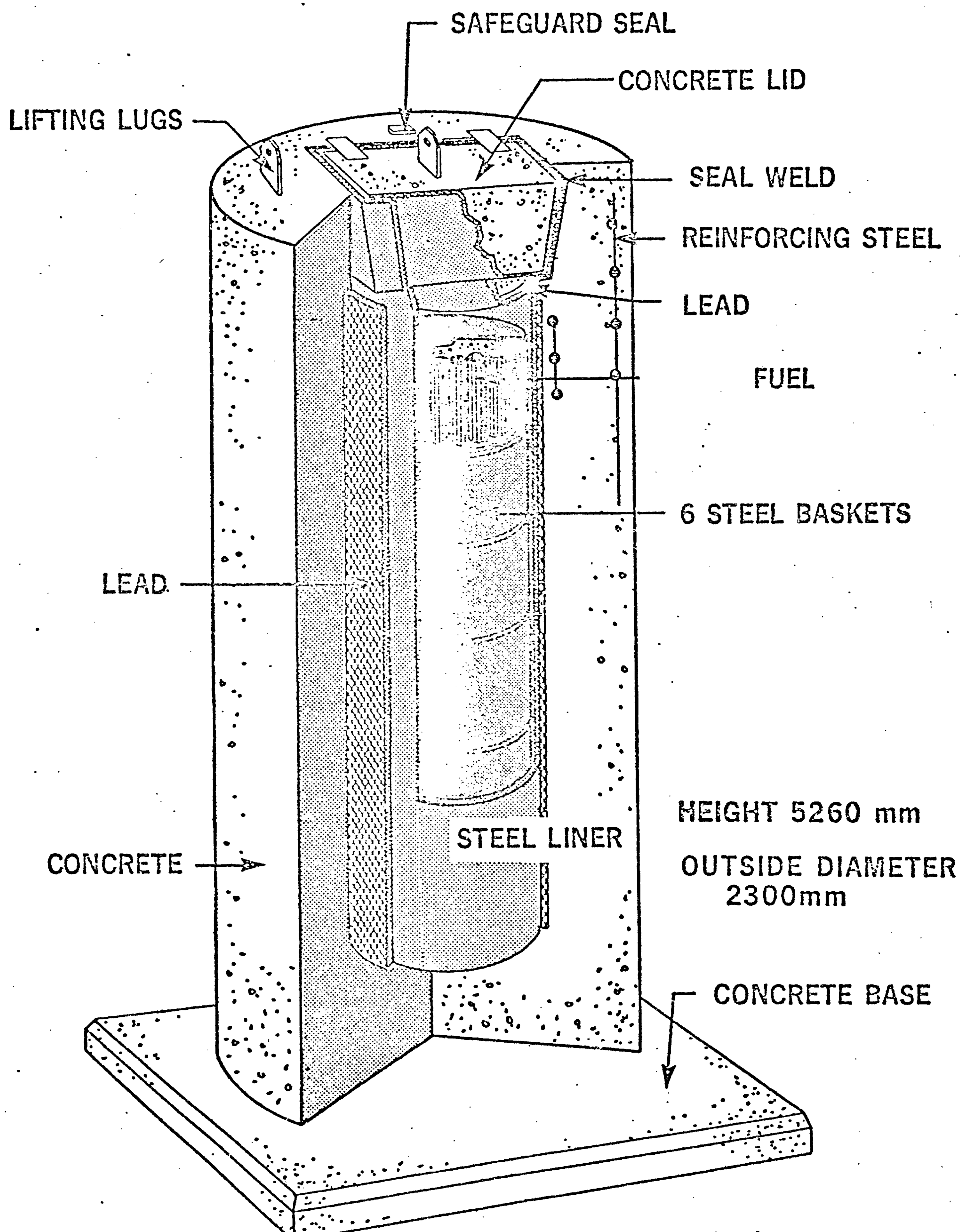



FIGURE 3: Maximum Canister Temperature Versus Soil Conductivity for the Dry Well Concept

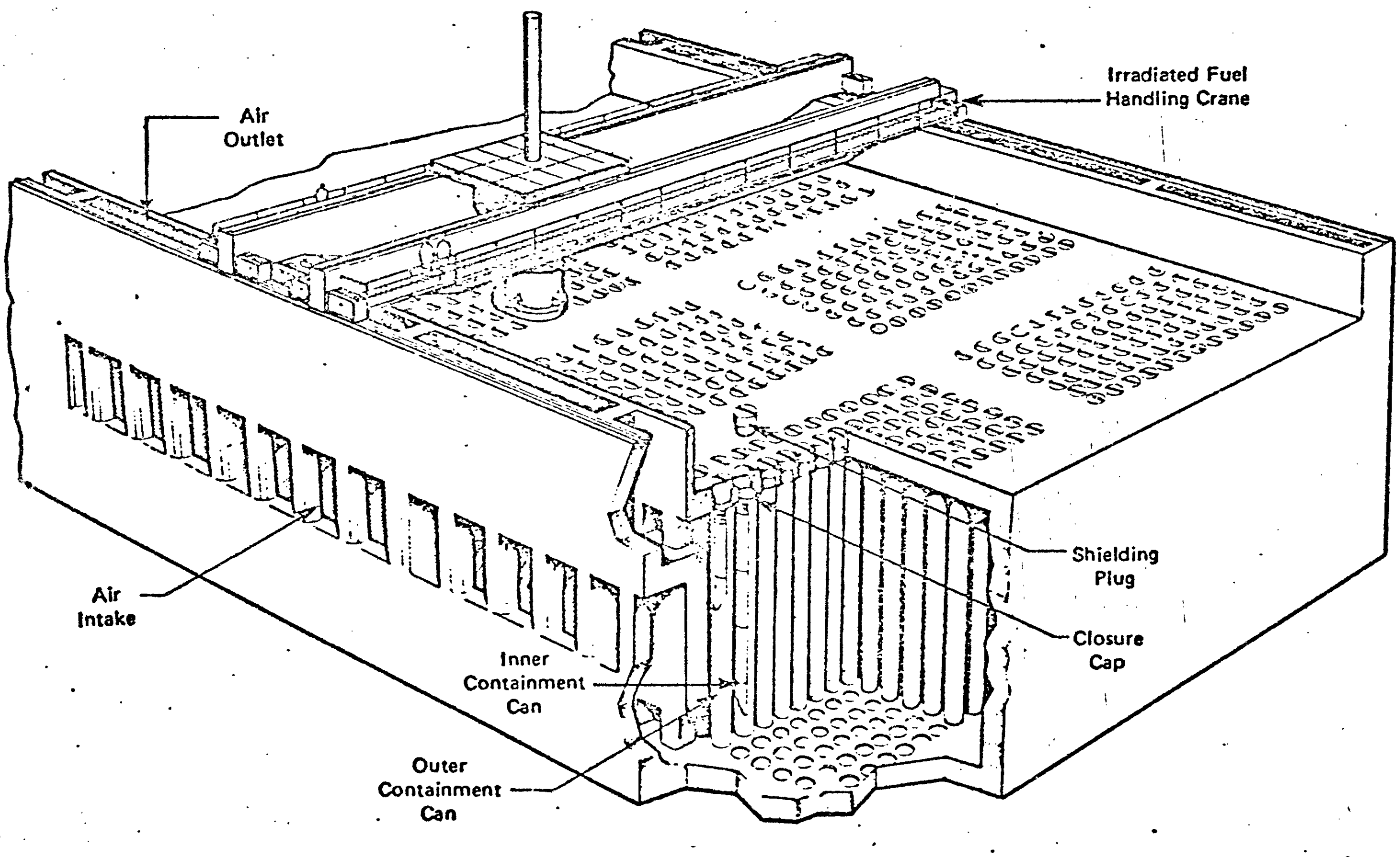
| | | |
|---|---|---|
| Assumptions: | Canister Thermal Power | = 0.964 kW |
| | Canister Spacing | = 25 feet |
| | Canister Internal Diameter | = 13 inches |
| | Canister Atmosphere | = Helium |
| | Canister Emissivity | = 0.3 |
| | Fiel Rod Emissivity | = 0.4 |

FIGURE 4
FUEL STORAGE CANISTER



# FIGURE 5

## FUEL STORAGE CONVECTION VAULT



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1. Management of Commercial High-Level and Transuranium-Contaminated Radioactive Waste, U.S. AEC environmental statement, WASH-1539, draft. Sept. 1974.

2. Interim Storage of Solidified High-Level Radioactive Waste, Panel on Engineered Storage, National Academy of Sciences, 1975.

3. A.B. Johnson, Jr., Behavior of Spent Nuclear Fuel in Water Pool Storage, BNWL-2256, Battelle Pacific Northwest Laboratories, Richland, WA, Sept. 1977.

4. To be precise, some zircalloy-clad and stainless steel clad fuel has been stored satisfactorily for periods up to 18 years and 12 years respectively; however, the bulk of the commercial high burnup LWR fuel and CANDU fuel currently in storage has been under water for five years or less.

   A possible exception which bears further study is the evidence of intergranular corrosion of stainless steel-clad Advanced Gas Reactor (AGR) spent fuel which has been exposed in core to temperatures in the range of 450-600°C. The apparent cause is sensitization of the stainless steel at these high temperatures. No evidence of this phenomenon has been observed with stainless-clad LWR fuel which has been exposed in core to temperatures in the range of 280-340°C. (A steel in which chromium carbide has precipitated at the grain boundaries, thus decreasing the chromium content in these regions, is said to be "sensitized.")

5. D.G. Boase and T.T. Vandergraab, "The Candaian Spent Fuel Storage Canister: Some Materials Aspects," Nucl. Tech., Vol. 32, pp. 60-71, Jan. 1977.

6. K.H. Henry and D.A. Turner, Storage of Spent Unreprocessed Fuel (SURF), RHO-SA-40, Rockwell INternational, Richland, WN, March 1978.

7. L.M. Richards and M.J. Szulinski, "Subsurface Storage of Commercial Spent Fuel," American Nuclear Society Topical Meeting on the Back End of the LWR Fuel Cycle, Savannah, Georgia, March 19-23, 1978, and M.J. Szulinski, private communication.

8. Typically, high-level waste solidified in borosilicate glass with a fission product content of $\sim$20% would have a heat rate of $\sim$6W/kg at 10 years of age; the heat rate of spent LWR fuel of the same age is $\sim$1W/kg.

9. M.M. Ohtu, The Concrete Cannister Program, AECL-5965, February, 1978.

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11. Report of the Task Force for Review of Nuclear Waste Management, US Department of Energy, DOE/ER-0004/D, February 1978, Appendix E, pp. 55-59

12. Handling and Storage of Spent Light Water Power Reactor Fuel, draft generic environmental impact statement, U.S. Nculear Regulatory Commission, NUREG-0404, Vol. 1,2, March 1978

13. Regional Nuclear Fuel Cycle Centers, Vol. II, Basic Studies, 1977, Report of the IAEA Study Project, Section 14

14. The concept of an energy island incorporating spent fuel storage reprocessing, waste disposal and fast breeder reactors has been extensively developed by W. Haefle and his associates. See; e.g., Nuclear Waste Storage and the Energy Island: A New Possibility for Action, International Institute for Applied Systems Analysis, Luxenburg, Austria, 1 October 1977

15. O. Marwah, Regional Storage of Spent Nuclear Fuel: The Indian Ocean Area, Program for Science and International Affairs, Harvard University, March, 1978