July 19, 1999 SECY-99-187

FOR: The Commissioners

FROM: William D. Travers /s/
Executive Director for Operations

SUBJECT: INFORMATION PAPER ON THE VIABILITY OF ENTOMBMENT AS A DECOMMISSIONING OPTION FOR POWER REACTORS

PURPOSE:

To inform the Commission of the staff's assessment of the viability of the entombment option for decommissioning power reactors.

BACKGROUND:

This paper is in response to a Staff Requirements Memorandum (SRM) on SECY-96-068 that addressed Direction Setting Issue (DSI-24) on decommissioning power reactors. The April 3, 1997, SRM requested an analysis from the staff on whether they view entombment as a viable option for decommissioning power reactors and how the Commission has dealt with previous licensee entombment option requests. The SRM stated that if the staff concludes that entombment is not a viable option, the staff should describe the technical requirements and regulatory actions that would be necessary for entombment to become a viable option for decommissioning. Further, the staff analysis should include an analysis of the resources involved, potential savings on decommissioning costs, and vulnerabilities.

SECY 98-099, dated May 4, 1998, contains a status report of the staff's review. That report summarized the present NRC regulatory position and requirements about the use of the entombment option. Based on a preliminary assessment of the efforts of Pacific Northwest National Laboratory (PNNL), the NRC staff indicated that consideration of entombment as a viable decommissioning option has merit. The staff further stated that rulemaking would be needed before this option could be treated as a generic alternative.

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Since the May 4, 1998 status report, the staff has completed a technical feasibility analysis of entombment as a decommissioning option. PNNL provided the basis for the staff analysis. Neither the staff analysis nor the PNNL assessment included any information from external stakeholders, such as the States and Low Level Waste Compacts. The rest of this paper discusses approaches, findings, and recommendations resulting from this analysis.

DISCUSSION:

In response to the Commission's request of April 3, 1997, the staff contracted with PNNL to evaluate the relevant technical issues associated with using the entombment option to decommission power reactors. A summary of the PNNL assessment and the staff's analysis of the PNNL assessment are provided in Attachment 1. The complete PNNL assessment is provided in Attachment 2.

The PNNL assessment considered realistic entombment scenarios that they believed were important from the perspective of regulatory significance to bound the entombment options contemplated by licensees. For those chosen entombment scenarios, PNNL addressed the suitability and concerns for choosing entombment configurations that can isolate the contained radioactive materials and the degree of institutional controls needed to ensure that these isolating properties are maintained. A fuller discussion of these scenarios and their regulatory implications was presented in SECY-98-099.

Regulatory Impacts and Issues:

Currently, there is a requirement that decommissioning be completed within 60 years of permanent cessation of operation (10 CFR 50.82(a)(3)) However, an alternative beyond 60 years could be approved, but only if shown necessary to protect public health and safety. Examples of special circumstances to be considered by the Commission in approving an alternative are the presence of other nuclear facilities at the site or a lack of waste disposal space. These exceptions are treated case-specifically and require Commission approval. In order for entombment to be treated on an equal basis with the prompt and deferred dismantlement options cited in the original 1988 rule, the 60-year time limit would need to be revised. In the Supplementary Information to the 1988 rule, periods on the order of 100 years were identified as being acceptable for private institutional control.

A second potential revision concerns the requirements in Subpart E of 10 CFR Part 20 for license termination under restricted release conditions. In the existing requirements, the possibility of the restrictions failing is considered and limits are placed on the maximum doses permitted to members of the public should failure occur. Under an entombment scenario, the most likely source of exposure would be inadvertent slow leakage of contaminants from the structure. A less probable scenario would be one where an intruder would unintentionally gain access to the radioactive materials inside the structure. The latter scenario is considered unlikely because the grouting and reinforced concrete structures used in the entombment (with the contaminants placed under ground) should deter or prevent inadvertent intrusion. As noted in SECY-98-099, it would be necessary to develop requirements to demonstrate that entombment system failure would be extraordinarily unlikely and would not result in doses greater than those permitted in 10 CFR Part 20, Subpart E. Also, a supplemental environmental impact statement is required, especially for those decommissioned reactors for which license termination occurs promptly and most of the radioactive inventory remains onsite. Some analysis was already performed for selected situations in the existing Generic Environmental Impact Statement, NUREG-0586.

The types of revisions just discussed are consistent with the basis for the current 10 CFR Part 20, Subpart E license termination rule. In that rule, the use of engineered features to reduce doses is discussed and is similar, when generalized, to the use of an entombment approach for providing isolation of the radioactive waste. Moreover, a reliance on institutional controls is also paralleled in the 1997 rule, where the requirement for periodic rechecks enters into the license termination. This issue is similar to that recently addressed by the Commission in a June 3, 1999, SRM related to the West Valley Decommissioning criteria proposed in SECY-98-251 and SECY 99-057. The Commission chose to apply 10 CFR Part 20, Subpart E. For the entombment approach to be viable, the Commission would need to decide whether there is reasonable assurance that without institutional controls intruder barriers would remain effective so that doses to the average member of the critical group would not exceed the dose limit criteria specified in the license termination rule. If the Commission decided to amend 10 CFR Part 20 as discussed above, there would be greater reliance on engineered barriers than there is today. This aspect has been discussed throughout this paper and in the earlier status report, SECY-98-099.

A significant issue is whether Greater Than Class C (GTCC) wastes can be entombed. GTCC waste contains very long-lived activation products, such as Nb-94. This waste is confined to small areas in the reactor vessel that had been subject to neutron irradiation over long periods of time. The lower plenum of a PWR is an example. This GTCC waste typically consists of about 11 cubic meters of material and it can be readily removed and shipped offsite. Because the GTCC waste is confined and results in highly radioactive materials, its removal can result in large radiation doses to the workers. Leaving these materials in the reactor vessel, as part of the entombed materials, could avert these occupational exposures. The GTCC nuclides and other activated radioactive contaminants (e.g.,Co-60) are intrinsically part of the reactor vessel steel materials. For most situations, these entombed steels will degrade very slowly, even if there is water leakage into the containment. Moreover, for the unlikely possibility of water breaching the contaminated materials containment structure, where the slow dissolution of the steels occur, the removal mechanism enabling the water to leave the containment is most likely to be diffusion. Diffusion is typically a very slow process. Finally, if the contaminants leave the containment they must still move through the surrounding soil media. At that time, it is likely that nuclide specific absorption would occur and residual contamination would also be diluted, thereby reducing the potential exposure. Therefore, the off-site dose to the public from these GTCC, very long-lived activation products, is expected to be very small.

From a technical perspective, isolation of GTCC radioactive materials that have been entombed appears to have realistic possibilities. However, Section 3(b)(2) of the Low-Level Radioactive Waste Policy Amendments Act of 1985 (Low-Level Waste Act) requires that GTCC waste resulting from NRC-licensed activities be disposed of in a facility licensed by the NRC. Under NRC's regulations, in the absence of specific requirements in 10 CFR Part 61, GTCC waste "must be disposed of in a geologic repository as defined in 10 CFR Part 60 ... unless proposals for disposal of such waste in a disposal site licensed pursuant to this part are approved by the Commission" (10 CFR 61.55(a)(2)(iv)). Thus, under both the statute and NRC's regulations, disposal of GTCC waste must be under license. In the case of the reactor entombment option, as with the other decommissioning options, the satisfactory completion of decommissioning is license termination (with or without restrictions). Therefore, even if a technical solution for leaving the GTCC waste in the reactor could be implemented, use of an entombed reactor structure for the disposal of GTCC wastes would require amendment of the Low-Level Waste Act and of NRC's regulations. Any such statutory or regulatory changes would need to be coordinated with the Department of Energy (DOE). Under the Low-Level Waste Act, DOE has the responsibility to provide disposal capacity for commercially-generated GTCC wastes.

Based on the technical considerations discussed, it appears worthwhile to consider the issue of amending the Low-Level Waste Act for the power reactor GTCC entombment option case. In addition to the above issues, a number of other institutional and regulatory issues need to be addressed before a recommendation to pursue entombment further can be made. Key among these issues are:

- How many licensees are interested in entombment as a decommissioning option?

- Can institutional controls and intruder barriers remain effective over the time duration required for entombment? What is required for reasonable assurance of their effectiveness?

- Can any above-ground structure be used, or should all entombment structures be below ground?

- Would entombment be considered inconsistent with the Low Level Waste Act, in that entombed reactor sites would represent a proliferation of low level waste disposal sites?

In addressing these issues, input from stakeholders should be sought.

Potential Savings:

As noted in Attachment 2, the major cost for prompt or delayed decommissioning can be the cost of waste disposal. Clearly, if more waste is entombed, this cost can be significantly reduced. However, for some entombment scenarios, the cost of maintaining and monitoring the entombed facility for 100 to 300 years can offset some of the savings advantage (see Table 1, on Page 14 of Attachment 2). If an entombment scenario is being considered where, for most of the time after permanent cessation of operations the license is still maintained and the reactor remains in a safe storage condition, the cost of this storage, over time, is part of the cost of the entombment scenario. Thus, if this scenario is compared with that of a dismantlement scenario, then there can be a significant amount of cost saving offset. However, if this scenario is compared with entombment costs once the license is terminated, institutional costs should be minimal. That assumes that the entombment was properly performed and the required institutional controls were advantageously planned. For example, a major concern of entombment is that water might get into the containment. This could be monitored using complex and costly measurement schemes. Alternatively, a small TV camera could be placed in the lowest part of the containment and any water observed on the TV could be sufficient for assessing this concern in an inexpensive way.

The waste volumes requiring disposal in the entombment scenarios are reduced from the waste volume resulting from dismantlement by about a factor of ten. Most of the waste volume from all of the entombment scenarios arises from the post-shutdown deactivation activities. Even these reduced volumes could be essentially eliminated by proper treatment and packaging of the deactivation wastes for inclusion within the entombment. As a result, the entombment scenario costs are not very sensitive to the rates charged by the low level waste disposal facilities. Low level waste disposal costs of the dismantlement scenario can be the largest single cost element for that scenario. PNNL estimated that disposal comprised from 20% to 40% of the total scenario cost, depending upon charge rates at the disposal facility. NUREG-1307, Revision 8, permits cost estimation assuming direct licensee disposal at Barnwell. In 1998 dollars, this results in the waste cost comprising 75% of the total decommissioning cost.

Alternatively, NUREG-1307, Revision 8, permits direct licensee disposal of the waste to a waste processing vendor. This results in a substantial savings and disposal costs would comprise 55% of the total cost. However, additional onsite waste volume reduction above that used in NUREG-1307, Rev. 8, is possible through more efficient contaminant removal methods and denser packaging of metal scrap. For a PWR, this reduction can be approximately a factor of 3, and results in a waste disposal cost which is about 20% of the total decommissioning cost. Depending on the circumstances that exist at the time of waste disposal, even this last estimate of the waste disposal cost percentage could be further reduced.

CONCLUSION:

Based on the PNNL assessment, the staff believes that decommissioning a power reactor using the entombment option can be safe and viable for many situations (depending on the site-specific circumstances). Entombment could provide greater flexibility to licensees for best accommodating their situations. If the entombment were properly performed, the impacts on health, safety, and the environment should be small, as concluded in Attachment 2. Moreover, other industrial, non-radioactive risks involved in the removal and disposal of these wastes would be eliminated, such as those activities used in the removal, packaging and transport of waste. However, as discussed above, to implement this option as an alternative to other decommissioning options would require changes to regulatory requirements and guidance. In addition, there are many issues involving statutory, regulatory, technical, and implementation matters whose implications require further development. For example, the staff believes that, for entombment scenarios where the radioactive dose concerns remain over very long time periods, the feasibility of acceptance will depend on the industry and the Commission resolving policy and technical issues where long term reliance is required on intruder barriers over the 1000 year period specified in 10 CFR Part 20, Subpart E. Accordingly, the staff recommends that a broader perspective and more detailed assessment of these issues be pursued as a precursor to any recommendation on whether or not to pursue legislative, regulatory, and technical implementation of the entombment option.

Therefore, as the next step in considering the entombment option, the staff intends to conduct a workshop in the near future. The purpose of this workshop is to solicit stakeholder views on the technical basis, issues, and options for treating entombment on an equal basis with other decommissioning alternatives, such as SAFSTOR and DECON. After conducting the workshop,

we will provide the findings from the workshop to the Commission along with our recommendations on whether or not to pursue entombment further and any policy issues needing Commission attention.

original /s/ by

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Attachments: 1. Summary of PNNL Assessment and Staff Analysis.
2. PNNL Assessment," Viability of the Entombment Option for Decommissioning Nuclear Power Reactors"

Attachment 1

Summary of PNNL Assessment of Entombment and Staff Analysis

PNNL Assessment

The PNNL assessment, dated May 11, 1999, considered realistic entombment scenarios sufficient to bound the entombment options considered by licensees.

Briefly summarizing, the scenarios considered were for license termination under restricted release (i.e., institutional controls) that at the time of termination (i.e., entombment) contained different amounts and types of radioactive contaminants. In the first entombment scenario (Example 1), the initial contaminants to be entombed are permitted to decay to low activity levels while the reactor facility is still under license (safe storage). Then, the entombment is closed and the license terminated with restrictions, as required in the existing requirements (10 CFR Part 20, Subpart E). In Example 2, significantly greater radioactive contamination remains, compared with Example 1, at the time of license termination. The crucial difference between Examples 1 and 2, both starting with the same large amount of radioactive inventory, is that Example 1 requires a much longer licensed safe storage time before its license is terminated. On the other hand, Example 2 relies on the properties of the (engineered) entombment configuration and institutional controls to protect the public from potential radiation exposure.

An additional consideration occurs if the very long-lived radionuclides that were generated during power operations by neutron activation are also allowed to remain. These nuclides of concern, when found in sufficient concentrations in waste, are classified as Greater Than Class C (GTCC). GTCC waste is above the highest Low-Level-Waste (LLW) classification of contaminated waste, Class C, that is permitted to be disposed of in LLW facilities licensed under 10 CFR Part 61. For reactor waste, the dominant GTCC direct dose contributor is Nb-94 and has a half-life of 20,000 years. Inclusion of GTCC waste in the entombment greatly affects the duration for which the entombment integrity must be maintained. For example, without GTCC waste, the time for which isolation of the contaminated waste is of concern (starting from the reactor permanent cessation of operations) ranged from 100 to 300 years. This time would be thousands of years if GTCC materials were also entombed. In addition, these time intervals are greater than the generic 60 year time interval permitted in 10 CFR 50.82.

In the assessment, PNNL examined the technical viability of the entombment options by considering several factors: the physical and chemical properties of the radioactive materials expected to be entombed, the integrity of the entombing structure, the site-specific sorptive and hydro-geological properties of the surrounding media, and realistic assumptions about radioactive contaminant transport behavior that could result in public exposure. PNNL had previously examined power reactor entombment possibilities at various times in their previous decommissioning studies. In those studies, as in the current one, the radioactive materials were assumed to be placed in the Reactor Pressure Vessel (RPV) and, if necessary and depending on the type of contaminant, in other places within the containment building. All waste was assumed to be below grade to take advantage of such placement's inherent structural preservation capacities, enhanced strength and integrity of the structure, and better

properties for isolating the contaminants. Depending on what was being entombed, use of fill materials such as concrete, or grout, or radionuclide specific sorptive materials (e.g., clays) were also considered. Also, hazardous liquids and some contaminated materials were assumed to have been removed, and the dose to workers was reduced by assuming decontamination of the primary system and removal of the decontaminating solution. For example, the contaminant inventory listing in Table B-1 of Appendix B of the PNNL report shows very low values of the fission products (such as Cs-137), compared with the activation products (such as Co-60). This is discussed in the last footnote in Table B-1.

The PNNL methodology used to evaluate entombment was similar to much of the approach used in NUREG/CR-0130, "Technology, Safety and Costs of Decommissioning a reference Pressurized Water Reactor Power Station," and NUREG/CR-0672, which is the equivalent report for the BWR. Their approach was to (1) examine information from actual reactor entombments for comparisons with NRC's power reactor entombment considerations, (2) consider available information from systems and activities that have similarities, and (3) do a comparative analysis of significant component / system behaviors that are similar to the entombment options components / systems.

Using this approach, PNNL examined information from three small DOE reactors that were entombed about 30 years ago. These were the only reactors ever entombed in the US. PNNL described lessons learned from these DOE entombment activities that could be applicable if NRC licensed power reactors are ever permitted to be decommissioned using an entombment option. PNNL considered the adequacy of entombment structural design criteria and the construction practices used, and the performance of periodic surveillance and monitoring to verify containment of the radioactivity. PNNL also evaluated previous DOE analyses on the potential doses to individuals from the entombed contaminants for various entombment scenarios. The dose pathways considered by DOE were from either direct intrusion into the entombing structure or from transport of radioactively contaminated materials to the surrounding environment.

PNNL considered relevant information from DOE waste burial activities, such as contaminant isolation assessments, and the associated health and safety requirements imposed. Also, to gain further insights, they reviewed 10 CFR Part 61 LLW requirements and examined what aspects of these requirements could be used for power reactor entombment considerations. Finally it should be mentioned for purposes of clarity that the decommissioning options denoted in the Attachment by DECON, SAFSTOR, and ENTOMB are equivalent to the options for dismantlement and entombment that are referred to in this paper.

Staff Analysis of the PNNL Assessment:

The staff believes that the PNNL evaluation provides reasonable technical evidence that entombment of radioactive wastes within the reactor structure can be a viable option for decommissioning. If there are no limiting site conditions, such as a shallow water table, the license can be safely terminated with one proviso. The licensee must satisfactorily demonstrate that the entombed radioactive contamination will be effectively isolated and that the individual radiation doses (either directly or from transport through the entombment surroundings) will be

kept below the acceptable limits (10 CFR Part 20, Subpart E). If this is the case, the contained radioactivity can be sufficiently isolated so that health and safety is not compromised or the environment degraded.

While some limited entombment options might be permissible under the existing 10 CFR Part 20, Subpart E, and 10 CFR 50.82 the more general case would require amendments to these rules before entombment could be permitted. Issues that must be considered for such an amendment include: the requirements for the isolation assessment of the entombment configuration contemplated and the institutional controls (monitoring and maintenance) that would be employed to provide assurance that the intended restrictions would remain effective for the necessary time.

If the isolation failed (e.g., because of mechanical or structural failure or chemical degradation), it would be possible for the radioactive materials contained in the entombment to be removed. Removal of this material would result in the generation of additional material that is contaminated and might require costly offsite disposal. The major component of the new material is the material used to shield and contain the radioactive contaminants within the entombed structure (e.g., soil, grout, concrete). Therefore, it is important that the waste isolation system be carefully assessed and properly implemented to reduce the likelihood of any failure occurring.

The PNNL analysis used existing regulations as a benchmark for performing their assessment. However, the Commission could consider using other risk management approaches (e.g., approaches used in the Resources Conservation and Recovery Act or DOE cleanup criteria) that place greater reliance on the durability of institutional controls.

Attachment 2

Viability of the Entombment Option as an Alternative for Decommissioning Commercial Nuclear Power Reactors

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The development and presentation of information, as well as any opinions and conclusions expressed in this report, are those of Pacific Northwest National Laboratory (PNNL). This report should not be construed by anyone as a means to represent the views of the Commission or the NRC staff. Any modifications to this report made by the NRC staff were solely for purposes of clarification or correction of information on NRC rule requirements. Moreover, any speculative statements made by PNNL concerning possible public reactions that might occur as a result of future rulemaking activities were removed. The purpose of this PNNL report is to serve as a technical information document, and be used by the NRC as a precursor for rulemaking considerations. The issue of public reaction to possible rulemaking requirements should be considered at the time when such rulemaking is being considered.

Introduction

In the years since the U.S Nuclear Regulatory Commission (USNRC) was established to regulate the production and use of radioactive materials and the construction/ operation/ decommissioning of nuclear power reactors, the USNRC has generally opposed on-site disposal (entombment) as a decommissioning alternative. This opposition was based upon analyses and comments on the Generic Environmental Impact Statement (EIS) on Decommissioning of Nuclear Facilities (NUREG-0586) and the Supplementary Information to the subsequent Decommissioning Rule [FR 24016, 1988]. The USNRC had a concern that the entombment enclosure might fail and allow the contained radioactivity to escape into the environment before it had decayed sufficiently to avoid unacceptable radiation doses to the public. Also, when comparing the impacts between the decommissioning alternatives of prompt or differed dismantlement and entombment, the deferred dismantlement decommissioning alternative appeared to offer comparable or better health and monetary results, and with much less potential risk to the public from direct exposure or transport of the radioactive contaminants to the biosphere. Part of the considerations for this last deduction was based on the readily available waste disposal accommodations and low cost of waste disposal at the time of development of the information base on decommissioning. This situation no longer is the case. Another concern focused on the effective creation of a number of low-level radioactive waste (LLW) disposal sites around the country, rather than a much more limited number of centralized disposal site facililities licensed under 10 CFR Part 61 (assuming that the Greater than Class C (GTCC) waste was not part of the inventory of materials for entombment. As part of these waste disposal considerations, a concern focused on the potential need for regulatory attention and institutional control of the entombed facility for a very long time, equivalent to the control times for present-day LLW disposal facilities. If GTCC waste was considered, there was potential need of even much greater time than that for the LLW disposal facility. As a result, while entombment was not specifically precluded by USNRC as a decommissioning alternative, the regulations and interpretations thereof allowed consideration of entombment as a decommissioning alternative only when necessary to protect public health and safety (i.e., when there was no space available in existing LLW disposal facilities).

With the issuance of the Rule on Radiological Criteria for License Termination [FR 39058, 1997], a nuclear facility license can now be terminated under either unrestricted use or restricted use release conditions. The restricted use release possible under the new rule allows the presence of residual radioactivity on sites whose licenses have been terminated, provided that certain constraints on annual radiation dose to the public can be met, and has generated renewed interest in the viability of entombment as a decommissioning alternative for nuclear power plants. Also, the escalation of the cost of LLW disposal has continued in recent years and is expected to increase even more rapidly if and when any of the planned regional Compact LLW facilities come into service, making the disposal costs the largest single element in decommissioning costs. Subsequently, several reactor licensees have requested that the USNRC review its previous positions regarding the entombment alternative.

The purpose of this report is to consider on-site disposal (entombment) of nuclear power reactors, and to examine the criteria, information base and the conditions under which entombment would be a viable decommissioning alternative in today's regulatory environment. The report begins with a brief summary of the information and conclusions presented in this report. The summary is followed by a review of past entombment experience in the U.S., an assessment of current USNRC decommissioning requirements (including the relevance of those requirements to the entombment issue), the aspects of these requirements that may preclude consideration of entombment as a decommissioning alternative, and how the requirements would need to be changed to make entombment a viable alternative. An important focus of this assessment is the ability or inability of an entombed power reactor to effectively isolate the radioactive contaminants considered for entombment from the biosphere so that the health and safety of individuals is not compromised. A discussion of postulated entombment

scenarios and the estimated costs, occupational radiation doses, and waste volumes is presented. Finally, the conclusions of the study and the path forward that to make entombment a viable decommissioning alternative are presented. Appendices containing more detailed discussions of the previous reactor entombments in the U.S., isolation assessments for entombment, corrosion issues related to reactor entombment, and radioactivity transport through soils in the environment are also attached.

Summary

Historically, only three small power reactors have been entombed in the U.S. Those facilities have been entombed for nearly 30 years, and no adverse effects on the public have been detected during this extended surveillance period. A number of facilities, which were once part of the federal government's nuclear weapons program and are located on federal property, are presently being considered for entombment (among other possibilities) (e.g., the waste tanks at Hanford and Savannah River sites. Other facilities in the weapons program complex may also be eventually entombed. Thus, there is past precedent for entombment of power reactors and entombment is being considered as a possibility (at least within the USDOE complex).

The basic regulatory framework to support entombment will require some modification, even with the issuance of the rules on Radiological Criteria for License Termination (1997), and Decommissioning of Nuclear Power Reactors (1996). The current constraint on the duration of a decommissioning action to 60 years or less would have to be revised to permit control of the facilities for up to 130 years or more, to allow the shorter-lived radionuclides to decay to insignificant levels, and significantly longer when GTCC materials are included, assuring that off-site radiation doses would be within regulations and that inadvertent intruders would receive little or no radiation dose.

Retention of the highly activated reactor vessel internals within the entombment enclosure presents some interesting problems. If the activity levels of any of that material exceeded Class C, as defined in 10 CFR 61, then the very long times required for control of the facility would have to be taken into account in the assessment of the entombment systems ability to effectively isolate this type of radioactive contaminant so that unacceptable health consequences result. For example, a potentially dominant dose contributor of such GTCC materials, when present above a minimum Curie concentration (below which it is considered as below the GTCC classification) is Nb-94. This radionuclide has a half-life of 20,000 years and is a gamma dose contributor that can result in a direct dose to someone inside the reactor pressure vessel (RPV) of several Rem/hr. Of course, entombment of GTCC materials would also need consideration in a required supplemental Environmental Impact Statement on entombment permissibility (EIS). Removal of the highly activated internals prior to entombment closure would greatly simplify regulatory modifications needed to facilitate entombment and, if rulemaking was considered worthwhile, likely result in a significantly faster completion. Then, if still considered worthwhile, the entombment permissibility of the GTCC materials could be undertaken.

The key element in obtaining approval for entombing a reactor is the isolation assessment of the enclosure and its contents, to show that the radionuclide release rates from the enclosure would be sufficiently small to satisfy the criteria set forth in the Radiological Criteria for License Termination Rule, when amended to permit entombment consideration inclusion. For entombment reasonably soon after reactor shutdown (< 50 years), the large inventory of short-lived radionuclides in the plant, which is more mobile than the long-lived activation products, must be included in the isolation assessment, and the measurements necessary to determine that component of the source term would be difficult and complex. The principal radionuclides of concern (^{94}Nb and ^{59}Ni), while very long-lived (20,000 year and

80,000 year half-lives), are generally contained in a stainless steel matrix which tends to resist corrosion. The release of these radionuclides into the environment will be very slow, even if the materials are immersed in water. Thus, it appears that many (if not all) reactors could probably satisfy the performance objectives set forth by the amended Rule. However, an in-depth assessment of the enclosure's isolating ability at various decay times following reactor shutdown would be necessary to determine when the enclosure could be permanently sealed, for each individual reactor considered for entombment.

A person inadvertently intruding into an entombed reactor could conceivably receive a significant radiation exposure if the intrusion occurred early in the lifetime of the enclosure (i.e., within 50 - 60 years following reactor shutdown, before the residual ^{60}Co and other short-lived radionuclides have decayed to insignificant levels. Careful attention to closure design and strength would be necessary to assure against an early intrusion. An intrusion after 130 years would not produce any significant radiation exposure so long as the activated reactor vessel components containing ^{94}Nb are properly secured within the enclosure, perhaps by grouting those components within the vessel. In addition, the vessel, its components, and the surrounding biological shield could all be grouted together, creating a monolith that could not be breached by an inadvertent intruder. Of course, a determined and well-equipped intruder could probably breach any enclosures, but that action would hardly be inadvertent.

The range of possible entombment scenarios is bounded by Immediate Entombment, with closure occurring shortly following reactor shutdown, and Delayed Entombment, with closure occurring about 120 years following reactor shutdown. Analyses have shown that (assuming the costs of nuclear insurance and facility security can be reduced to reasonable levels) the Delayed Entomb scenario will result in the least cost (both constant dollar cost and present value cost), and the least occupational radiation dose of any of the decommissioning alternatives by factors of ranging up to about 3.

In conclusion, it appears that entombment is a viable technical alternative for decommissioning of nuclear power reactors. Relatively minimal revisions would be needed to existing USNRC regulations to facilitate entombment.

Entombment Experience in the U.S.

In the more than 50 cases of reactor decommissioning in the United States since 1954, only three installations utilized the onsite disposal strategy. Those three reactors were relatively small demonstration plants built by the U.S. Atomic Energy Commission in the early days of nuclear power development, namely:

- The Hallam Nuclear Power Facility,
- The Piqua Nuclear Power Facility,
- The Boiling Nuclear Superheater Power Station (BONUS).

These reactors were entombed in the 1969/1970 time period, prior to the creation of the USNRC and therefore were not governed by USNRC regulations.

While the experience base for nuclear reactor entombments in the U.S. (or throughout the world for that matter) is not extensive, there are lessons that have been learned from the entombment of these three reactors that would be applicable to entombment of USNRC-licensed commercial reactors:

- entombed nuclear reactor structures will contain the residual radioactivity as long as the entombed structure is adequately designed to appropriate criteria and properly constructed. Some reactor core components and all bulk chemically hazardous materials (and unirradiated and spent fuel) should be removed and disposed of off-site,

- penetrations into the reactor building should be plugged and sealed and the entire entombed reactor structure should be sealed against intrusion and weatherproofed, and

- periodic surveillance and monitoring should be conducted to verify containment of radioactivity.

In addition, disposition of eight former plutonium production reactors owned by the U.S. Department of Energy (USDOE) and located on the Hanford Site in the State of Washington have been considered in an Environmental Impact Statement (EIS) [USDOE, 1989] whose options included onsite disposal (partial dismantlement and subsequent entombment). These reactors had characteristics rather similar to commercial power reactors, as listed below:

- located near a major river,
- located in an arid region having a low population density,
- relatively large residual inventories of long-lived radioisotopes (e.g., ^{14}C and ^{63}Ni), and
- residual radioactivity inventories posing possible consequences to intruders (i.e., ^{94}Nb).

While the in-place disposal option was not selected in the Record of Decision as the preferred alternative for these reactors, the environmental impacts for this and the other options considered in the EIS did not offer a strong basis for differentiation and selection among the options. The preferred option was ultimately selected based on other factors, including the results of the public hearing process. With regards to the entombment of USNRC-licensed commercial reactors, the results of the EIS do demonstrate that, under some circumstances, the environmental impacts of entombment of large reactors with significant residual source terms are not significantly different from the environmental impacts of dismantlement alternatives.

Also within the USDOE complex, consideration has been given to the entombment of the waste tanks at the Hanford site [USDOE, 1987] (and possibly at the Savannah River site), which currently contain the highly radioactive liquid wastes from spent fuel reprocessing, after those wastes have been removed. One scenario considered anticipated that any residues remaining in the tanks would be grouted in place, with the tanks backfilled with sand and gravel, sealed, and controlled within the federal reservation.

More discussions of these existing and possible entombment cases is provided in Appendix A.

Regulatory Considerations

The USNRC recently issued two new rules that deal directly with the requirements for decommissioning of licensed nuclear facilities. These are :

- Decommissioning of Nuclear Power Reactors [FR 39278, 1996], and
- Radiological Criteria for License Termination [FR 39057, 1997].

In addition, existing regulations contained in Title 10, Code of Federal Regulations, Part 61 (10 CFR 61) govern operation and closure of LLW disposal facilities. Although license termination of a power reactor by decommissioning it using the entombment option would not come under the requirements of the 10 CFR 61rule requirements (but, instead, by the two rules above preceeding this rule), it is instructive to compare which similarities and differences of the Part 61 rule could apply to entombment upon final closure. The effects of these rules and regulations on the viability of reactor entombment are discussed in the following three subsections.

Decommissioning of Nuclear Power Reactors

The Decommissioning Rule [FR 39278, 1996], and its 1988 predecessor [FR 24018, 1988], contains a limitation on the time after shutdown (60 years) by which the licensee must have completed decommissioning actions [10 CFR 50.82(a)(3)]. Selection of this period duration was based largely upon the decay rate of the dominant radioactive species (^{60}Co) that contributed to the occupational radiation dose received during decommissioning operations at a reactor, and was reasonably optimal for minimizing occupational radiation dose and long-term safe storage costs.. This 60-year limitation had all but precluded consideration of entombment as an option for decommissioning of nuclear power reactors because decommissioning was defined prior to 1996 as removing a facility "safely from service and reduce residual radioactivity to a level that permits release of the property for unrestricted use and termination of the license,"and because the significant residual radioactive inventory in an entombed reactor generally made it not possible to release the site for unrestricted uses within 60 years after shutdown. However, the 1996 Decommissioning Rule also redefined decommissioning to allow the "release of the property under restricted conditions and termination of the license,"so long as the annual radiation dose to members of the public were sufficiently small (see next subsection on Radiological Criteria for License Termination). An assessment of the ability of an entombed nuclear power reactor to satisfy the radiological criteria for site release within the 60-year limitation is presented in the section titled Entombment Isolation Assessment, below.

Radiological Criteria for License Termination

The charter of the USNRC is to protect the health and safety of the public against excessive exposure to radiation arising from licensed nuclear activities. Decommissioning of a licensed facility is the final step in the cessation of nuclear activities at that facility and, under the 1997 Rule governing radiological criteria for license termination [FR 39057, 1997], to terminate a nuclear license it is necessary to show by demonstration and/or analyses that the dispersal of the residual radioactivity at the site will not result in a radiation exposure to any member of the public arising from living or working on the previously licensed site exceeding 25 mrem/yr (distinguishable from background), for unrestricted use of the site [10 CFR 20.1402]. For restricted use of the site, higher levels of residual radioactivity are permitted, provided that legally enforceable institutional controls are established that will limit the radiation dose to a member of the public to 25 mrem/yr (distinguishable from background) or less [10 CFR 20.1403(b)]. However, residual radioactivity at the site must be sufficiently small that, in the event that the institutional controls are no longer in effect, the radiation dose would not exceed 100 mrem/yr (distinguishable from background) or 500 mrem/yr under unusual circumstances [10 CFR 20.1403(e)]. The potential ability of an entombed nuclear power reactor to satisfy these performance objectives is discussed in the section titled Performance Assessment, below.

Licensing Requirements for Land Disposal of Radioactive Wastes

Decommissioning a power reactor using the entombment option for license termination should come under the requirements of the Decommissioning Rule and the Radiological Criteria for License Termination Rule discussed above. Additional technical insight can be gained by observing which parts of the 10 CFR 61 license requirements for LLW disposal would have relevance for regulating entombment of nuclear power reactors when final entombment closure has been completed. Such comparisons are made by considering which exceptions to the 10 CFR61 licensing process would have to be made if entombment were regarded as a LLW licensing process. The various factors that influence the ability of an entombed reactor facility to satisfy these requirements are discussed briefly below.

- Entombed reactor sites will not have adhered to the licensing process wherein "the potential applicant goes through a process of disposal site selection by selecting a region of interest, examining a number of possible disposal sites within the area of interest and narrowing the choice to the proposed site" [10 CFR 61.7(c)(1)]. Exceptions to this licensing process would

have to be made for entombed reactors.

- Because the criteria for siting a nuclear power plant (i.e., being located in relatively close proximity to a large body of surface water) is inherently different than that for LLW disposal sites, current disposal site suitability requirements for land disposal of LLW cannot be satisfied by many potential entombed reactor sites. Specifically, requirements precluding LLW disposal in a 100-year flood plain, coastal high-hazard area or wetland [10 CFR 61.50(a)(5)] or in the zone of fluctuation of the water table [10 CFR 61.50(a)(7)] are not necessarily compatible with existing reactor site characteristics. Thus, special exceptions to existing 10 CFR 61 requirements would be necessary to permit dealing with an entombed reactor under 10 CFR 61 following final closure. Further study is required to determine specifically which nuclear power reactors may or may not be able to satisfy the disposal site suitability requirements.

- The performance objectives for USNRC-licensed LLW burial grounds are that the radiation dose to any member of the public must not exceed 75 mrem/yr to the thyroid or 25 mrem/yr to any other organ or to the whole body. The ability of an entombed reactor to satisfy these objectives is discussed in the section titled Performance Assessment, below.

- A requirement for the operation and closure of near-surface disposal facilities is that wastes designated as Class C "must be disposed of so that the top of the waste is a minimum of 5 meters below the top surface of the cover or must be disposed of with intruder barriers that are designed to protect against an inadvertent intrusion for at least 500 years" [10 CFR 61.52(a)(2)]. Where site conditions prevent deeper disposal, intruder barriers such as concrete covers may be used [10 CFR 61.7(b)(5)]. While entombed power reactor sites may not meet the 5-meter criteria, proper design and implementation of engineered barriers during the closure of the facility should allow the entombed structure to satisfy the requirements for the intruder barrier. Realistic and acceptable intruder scenarios need to be developed and evaluated for possible entombment designs.

- For LWRs that have operated for their full licensed lifetime (40 years), there will be significant portions of the reactor vessel internals whose activity levels for the long-lived ^{63}Ni, ^{14}C, ^{94}Nb, and ^{59}Ni radionuclides will equal or exceed Class C levels. Current regulations specify that the Greater-Than-Class C (GTCC) material is not generally acceptable for near-surface disposal, unless specifically approved by the USNRC on a case-by-case basis [10 CFR 61.55(A)(2)(iv)], and generally must be placed into a geologic repository licensed by USNRC. Presently, there is no geologic repository licensed by the USNRC for GTCC disposal. The general belief is that the federal high-level waste (HLW) repository will eventually be designated to receive GTCC wastes, but that repository is not expected to be in service for at least 15 years, and the repository operator (USDOE) has not yet agreed to receive GTCC material other than that which is an integral part of spent fuel assemblies.

Entombment of reactors that have had GTCC waste removed would appear to be more feasible from a regulatory perspective of 10 CFR 61 LLW disposal since near-surface disposal of Class A, B, and C wastes is already permitted under existing regulations and is a well-demonstrated safe mode of disposal for these classes of wastes in the U.S. Clearly, LLW within an entombed reactor facility could readily be made to meet both the minimum packaging and stability requirements for waste form as defined in 10 CFR 61.56. However, exceptions to those requirements related to site selection and disposal site suitability would be needed to permit reactor entombment under 10 CFR 61. Thus, for such a preference, it may be necessary for USNRC to require the GTCC material to be packaged for repository disposal and to be stored

with the reactor's spent fuel until a repository is available, thereby expressly excluding GTCC material from an entombed reactor.

However, license termination by decommissioning using the entombment option should be regulated by using the requirements of the Radiological Criteria for License Termination Rule, and there can realistically be situations for consideration of entombments containing the GTCC materials where the isolation assessment acceptably demonstrates that the license termination restricted release dose values will be within the regulatory requirements. Therefore, the possibility of entombments containing GTCC materials should also be a serious consideration.

- According to the requirements specified in 10 CFR 61.59, institutional control of the entombed reactor would have to be maintained for up to 100 years following closure, including physical control of access, environmental monitoring, periodic surveillance, and minor custodial care. Associated with the institutional control are the cost of performing the activities identified above, and funding to cover those costs must be assured during the period of institutional control [10 CFR 61.62]. If a reactor were entombed immediately after shutdown, that control period would be approximately 130 years. If a reactor were entombed following an extended safe storage period, it would seem logical to give the licensee credit for the surveillance and monitoring conducted during the storage period and shorten the institutional control period after entombment closure, such that the total control period (safe storage and entombment) would remain approximately 130 years following reactor shutdown.

Entombment Isolation Assessment

The key evaluation in determining the viability of entombment is the assessment of the isolation of the enclosure in retaining the residual radioactivity and the analysis of the doses to the public arising from dispersal of the contained radioactivity throughout the environs surrounding the enclosure over time. Factors important to the enclosure isolation assessment include:

- Radioactivity inventory present within the entombment enclosure
- Long-term physical integrity of the entombment enclosure structure and ability to exclude groundwater
- Physical and chemical forms and solubility rates of the individual radioactive species in various qualities of water
- Transport of the dissolved radioactive species through the entombment enclosure materials
- Dispersal of the dissolved radioactive species throughout the environment outside of the entombment enclosure structure.

Each of these factors, and the corresponding data for the reference PWR, are discussed in detail in Appendix B. While a specific isolation assessment focused on entombment of nuclear power reactors was not conducted for this study, inferences as to the likely outcome of an isolation assessment for a reactor are possible by comparison with analogs, for the residential/farm family and inadvertent intruder scenarios

commonly considered in performance assessments. These comparisons are discussed in the following subsections.

Residential/Farm Family

The analogs used here are LLW burial grounds located at two different USDOE sites, the Hanford Site located in Washington and the Savannah River Site located in South Carolina, having vastly different site characteristics. Appendix B contains a detailed discussion of the results of performance assessments conducted for LLW burial grounds at each of these sites. These performance assessments, for burial grounds at each of these sites, demonstrated that the USDOE performance objectives for long-term isolation could be met. The inference from the results of the performance assessments for these sites is (considering the differences in radionuclide inventory, site characteristics, and burial structure design relative to those for nuclear power plants) that it will likely be possible to show that entombment of many nuclear power reactors will meet stringent isolation objectives such as the 25 mrem/yr criteria in the radiological criteria for license termination rule for restricted release within 60 years after shutdown. These results also suggest that minimal monitoring will be required to verify entombment structure integrity, certainly much less than 60 years. These conclusions are based on the following observations:

- reactor building structures are at least equivalent to or significantly exceed the construction integrity and resistance to degradation required of the reinforced concrete vaults planned for LLW facilities,

- the residual inventories of radionuclides at an entombed reactor will be smaller than the inventories within LLW vaults (assuming that the GTCC material has been removed from the reactor) and those radionuclides will have less mobility than those evaluated for the LLW disposal vaults,

- the radionuclides of concern for entombed reactors are predominantly contained within a steel matrix and are released primarily via corrosion of that matrix (see Appendix C), whereas the radionuclides in the LLW vaults were contained within a grout matrix which is more susceptible to release via leaching of the radionuclides from within the grout.

- those radionuclides (such as ^{129}I and ^{99}Tc) that generally pose significant performance issues for LLW disposal are not present in significant quantities at an entombed reactor, and

- the USDOE performance objectives for long-term isolation of the radioactivity in the LLW vaults are essentially the same or at least similar to the criteria specified in the radiological criteria for license termination rule.

However, these beneficial characteristics may be somewhat offset by the following:

- the water table near many nuclear power reactor sites is much nearer to the surface than at the LLW burial grounds, which may increase the rate of degradation of the concrete structure and provide greater opportunity for water to infiltrate the structure

- most nuclear power plants are located much nearer to major rivers and population centers than are the LLW facilities considered, thereby increasing the potential for radiation dose to the population from this pathway
- several nuclear power plants are located close to sources of saltwater, which enhance corrosion rates of contaminated and activated steels and degradation of concrete structures. The saline

environment may cause the rates of radionuclide transport through the soil and environment to be greater than would otherwise be assumed.

One or more of these factors may ultimately be sufficient to prevent specific entombed reactors from meeting the required performance objectives. The sensitivity of isolation assessment results to residual source terms, site characteristics, and entombment structure designs needs to be determined.

Inadvertent Intruder

One major concern related to entombment is the possibility of inadvertent intrusion by members of the public into the entombment enclosure, with the resulting possibility of serious exposure to radiation during the intrusion and the possible transport of radioactive materials into the public domain, with subsequent exposure to larger segments of the population. The inadvertent intruder scenario generally provides the greatest challenge to being able to demonstrate that performance objectives for long-term isolation can be met. The LLW burial grounds discussed in Appendix B were shown to be able to meet stringent performance objectives. Each was designed with the specific intention of preventing inadvertent intrusion through combinations of engineered barriers and warning markers. Similar barriers and warning markers can be applied to entombed reactors.

For this study, an extremely conservative calculation was made to place an upper limit on the possible external radiation exposure hazard to an inadvertent intruder. It was assumed that an intruder resides within the reactor vessel (where the activated metals comprise the majority of the residual source term at the reactor). The radiation dose rate to such an intruder at 60 years after shutdown was estimated to be about 200 mrem/hr (1,800 Rem/yr). This radiation dose rate decreases to 100 mrem/yr after about 135 years and to 25 mrem/yr after about 160 years. These results suggest that an upper limit on the institutional control period should be 135-160 years, with the actual period being somewhat less for less conservative source term assumptions.

The 135-year institutional control period, coincidentally, is consistent with the results of an analysis by Nuclear Electric of the U.K., who has proposed a safe storage period of 135 years for their Magnox Reactors and Advanced Gas-cooled Reactors, with the decision to dismantle or entomb the stations deferred until the end of their storage periods [Brown, 1992]. The proposed technical approach in the UK is to entomb by grouting the internal void spaces, sealing up and capping over the reactor structure which is then covered with a mound of sea-dredged sand that is capped with revegetated top soil. A 10,000-year design life was specified for the mound .

The inadvertent intruder scenario could be dealt with in several ways. First, if the highly activated reactor vessel internals are removed for packaging, storage, and disposal, and institutional control of the enclosure prevents any intrusions for 130 years, the principal sources of radiation dose will have decayed to insignificant levels and no serious radiation exposures would occur thereafter. Second, if institutional control cannot be maintained for 130 years, the closures and other barriers on the entombment enclosure would have be more secure and a good knowledge of the residual short-lived radionuclides would be required (as a function of time after reactor shutdown) to determine when institutional controls could be terminated without increasing public risk. In either case, if the highly activated vessel internals (no GTCC material) are retained within the vessel, the reactor vessel interior could be filled with grout to prevent ready access to the interior of the vessel and its internals even if the vessel head were removed. This approach would physically prevent an inadvertent intruder from entering any high radiation dose areas. In both cases, the vessel could also be grouted within the confines of the biological shield, thus preventing any direct access to the interior of the shield or to the exterior of the vessel without first penetrating the shield and the grout. Considering the thickness of the

shields and the grouted spaces, access to the activated reactor vessel and its contained internals would be very difficult and certainly not inadvertent. The downside to this approach is that later retrievability of the vessel, if required, becomes significantly more difficult (but not impossible).

Postulated Entombment Scenarios and Estimated Costs

The two basic scenarios postulated for decommissioning via the entombment alternative are (for the example of the GTCC materials early removal):

- Immediate Entombment, wherein disassembly, packaging, and placement of radioactive materials within the entombment enclosure occurs immediately following reactor shutdown activities, and the enclosure is sealed and monitored for 130 years.

- Delayed Entombment, wherein the facility is placed into extended safe storage (100-120 years), followed by disassembly, packaging, and placement of residual radioactive materials into the entombment enclosure, and the enclosure is sealed and monitored for an additional 10 years.

Each of these scenarios is discussed in some detail in subsequent subsections.

Immediate Entombment

The postulated scenario for Immediate Entombment is comprised of the following actions:

- Immediately dismantle all auxiliary structures, package radioactive materials for
  - off-site disposal, or
  - placement within the entombment enclosure
- Immediately remove and package the reactor vessel activated internals for storage and/or off-site disposal
- Perform an in-depth inventory of radioactivity within the entombment enclosure
- Demonstrate that future releases from the entombed material will result in acceptable radiation doses to the public (performance assessment)
- Maintain monitoring and surveillance for up to 130 years following reactor shutdown.

In this scenario, reactor shutdown and deactivation is followed immediately by entombment. 'Immediate' in this case means after all spent fuel has been removed from the reactor vessel and, depending upon the physical layout of the facility, from the spent fuel pool, which could be completed between 1 to 5 years following final shutdown. The reactor support facilities are dismantled and the radioactively contaminated materials and equipment are packaged for either transport off-site for disposal at a licensed LLW facility or for transfer into the confines of the planned entombment enclosure. Some dismantlement of systems and equipment within the enclosure will probably be necessary to facilitate placement of the packaged materials from the support buildings. There may or may not be sufficient space within the entombment enclosure for all of the packaged contaminated materials, thus possibly requiring some off-site LLW disposal.

An in-depth inventory of the radionuclides remaining within the enclosure at the time of closure is necessary for the analyses needed to determine whether or not the future release rates (and associated

radiation dose rates) from the enclosure would permit license termination to be achieved. Some portions of the activated reactor vessel internals could have activities that exceed the Class C LLW classification and would require packaging and disposal in a federal geologic repository. The remaining carbon steel vessel and its stainless steel liner are also highly activated but will not exceed Class C levels, and so could remain within the entombment enclosure. For immediate entombment, it is necessary to consider the possible near-term release of the relatively short-lived surface contaminants such as ^{60}Co and ^{137}Cs over the first 130 years of entombment as well as the release over very long time periods of the very long-lived metallurgically-bound activation products such as ^{94}Nb and ^{59}Ni. Following entombment closure, environmental monitoring and site maintenance would be continued for up to 130 years following reactor shutdown, under the jurisdiction of the agency that controls the site during that period.

Much of the occupational radiation dose associated with immediate dismantlement would also be accumulated during immediate entombment, because the disassembly and packaging activities for the contaminated systems and equipment outside of the entombment enclosure would be essentially the same in both cases. Similarly, the labor costs for those activities would also be about the same in both cases. Within the enclosure, only those disassembly activities needed to clear out space for the packaged wastes from the other buildings would be needed, thus reducing both radiation dose and labor cost. The reactor vessel internals are removed and packaged for storage and/or off-site disposal while the vessel and the reactor fuel pools are still full of water, thus providing the shielding needed for these operations. Therefore, the radiation dose and costs associated with immediate disassembly, segmenting, packaging, transporting, storing and disposing of those very high activity materials would be incurred in both the immediate and the deferred scenarios. If the reactor were shut down well in advance of its licensed lifetime, as has been the case for most of the presently closed reactors, the internals would not have been activated to GTCC levels and may be retained within the reactor pressure vessel for entombment. The attendant radiation doses and costs for internals removal then would not be incurred in either scenario.

Because the presence of water within the entombment enclosure is a key factor in producing any releases of radioactivity from the enclosure, it may be appropriate to install a monitoring pipe into the lowest sump within the sealed enclosure. Periodic checking or remote monitoring of the sump for water intrusion and pumping of the sump if necessary to remove that water could provide assurance that the interior of the enclosure remained essentially dry and prevent dissolution and transport of radioactivity to the surrounding environs. Similarly, another entrant tube could be installed that could accept a television camera for scanning the interior of the enclosure, to provide visual assurance that severe corrosion and structural degradation was not occurring. The costs of utilizing these types of periodic or remote monitoring systems should be small compared with the costs of installing and monitoring numerous sampling wells in the environs of the enclosure.

Delayed Entombment

The postulated scenario for Delayed Entombment is comprised of the following actions:

- Immediately remove and package the vessel activated internals for storage and/or off-site disposal
- Defer all other dismantlement activities until the end of the safe storage period
- Dismantle all auxiliary structures, package radioactive materials for
  - off-site disposal, or
  - placement within the entombment enclosure
- Perform an in-depth inventory of radioactivity within the entombment enclosure

- Demonstrate that future releases from the entombed material will result in acceptable radiation doses to the public (performance assessment)

- Maintain monitoring and surveillance up to 10 years following entombment closure (130 years following reactor shutdown).

In this scenario, the activated reactor vessel internals are removed immediately and packaged for storage and/or off-site disposal, and the entire facility is placed into long-term safe storage to allow the shorter-lived radio-nuclides to decay to insignificant levels before the main disassembly and decontamination activities are performed. The principal contributor to radiation dose in the early years following shutdown (^{60}Co) will, after 50 years of safe storage, have decayed to about 3 orders of magnitude lower than shutdown levels and to about 7 orders of magnitude lower than shut down levels after 130 years of safe storage. Similarly, any ^{137}Cs contamination will have decayed by about a factor of 16 after 130 years. As a result, most of the contamination on systems and equipment will have decayed to unrestricted release levels after 100 to 130 years, and those volumes of material may be available for recycle and reuse, no longer requiring controlled disposal. Surface contamination throughout the facility will have also decayed to unrestricted release levels, virtually eliminating any need for surface decontamination within the facility. Thus, there would be little or no radiation dose associated with the disassembly and disposition of materials external to the entombment structure.

After 130 years of decay during safe storage, the principal radioactive components present in the facility will be the reactor vessel and its liner, and possibly some portions of the reactor biological shield. Retaining those materials within the entombment enclosure would further reduce radiation dose and labor and disposal costs for decommissioning.

While there are significant costs associated with the disassembly and packaging activities for the deferred entombment scenario, it is anticipated that at least part of these removal costs could be recovered from the scrap value of the removed non-radioactive materials. However, there will be additional costs associated with the period of long-term safe storage of this scenario. The magnitude of those costs will depend upon the extent to which monitoring, inspections, licensing, and maintenance costs can be minimized. Ideally, the facility would be placed into a passive state, with no full-time operating systems to monitor and maintain during safe storage. Also, monitoring for security purposes would be accomplished using remotely monitored electronic detection systems, eliminating the need for any full-time staff on-site. Inspections would be performed approximately annually, to check for intrusion of water into the enclosure and any leakage of contamination from the enclosure. Because the entombment enclosure remains open during safe storage, the interior of the enclosure can be checked periodically or remotely-monitored for any water in-leakage or other signs of containment deterioration, thus providing a history of enclosure structure performance prior to final sealing of the enclosure. With the vessel internals removed, the need for further monitoring of the enclosure would be quite limited because the shorter-lived radionuclides will have decayed to negligible levels after 100 to 130 years. If the vessel internals (no GTCC material) were retained within the sealed enclosure, the monitoring access pipes discussed previously could provide direct knowledge of any water intrusions and a way to remove that water, thus preventing dissolution and transport of the long-lived radionuclides to the enclosure environs during the monitored period. However, the ultimate test of entombment viability will be the isolation assessment analyses covering the contained radioactivity.

Estimated Costs, Waste Volumes, and Occupational Radiation Doses for Entombment

Earlier studies [Konzek, 1995; Smith, 1996] have shown that there can be some small cost reductions and some large waste volume reductions achieved, and there can be some significant occupational radiation dose reductions achieved by implementing entombment. These earlier studies have been revisited to examine the likely costs, waste volumes, radiation doses, and years of institutional control that would be associated with the original scenarios for DECON, SAFSTOR1, and Immediate ENTOMB, as well as with a new scenario, Delayed ENTOMB, which combines long-term safe storage with subsequent entombment. The Immediate ENTOMB scenario is also evaluated for a worst-case post-closure monitoring and surveillance period that extends to 130 years following reactor shutdown, and includes an evaluation of the effect of obtaining the much-reduced security and insurance costs during the latter period that were suggested and evaluated in Konzek [1995]. SAFSTOR1 is also evaluated for the reduced security and insurance costs. Similarly, the Delayed ENTOMB scenario is evaluated for the original and for the reduced security and insurance costs. The results of these evaluations are presented in Table 1.

From the table, it can be seen that the waste volumes requiring disposal in the entombment scenarios are reduced from the waste volume arising from DECON by about 1 order of magnitude. Most of the waste volume from the entombment scenarios arises from the post-shutdown deactivation activities, which is common to all of the scenarios. Even these reduced volumes could be essentially eliminated by proper treatment and packaging of the deactivation wastes for inclusion within the entombment enclosure. As a result, the entombment scenario costs are not very sensitive to the rates charged by the LLW disposal facilities, while the LLW disposal costs of the DECON scenario are the largest single cost element for that scenario and may comprise from 20% to 40% of the total scenario cost, depending upon charge rates at the LLW disposal facilities.

Table 1. Principal Elements in Decommissioning Analyses for Several Alternatives

	Cost (millions of 1993$)		Radioactive Waste Volumes (m^3)		Radiation Dose	Institutional
Alternative	Constant $	Present Value $	LLW	GTCC	(Person-rem)	Control (years)
DECON	133.3[(a)]	108.4	8,246	11	953	8.6
Immediate ENTOMB	162.1[(a)]	103.3	913	11	803	60
	251.0[(b)]	107.7				130
	143.9[(c)]	87.5				130
SAFSTOR1	174.0[(a)]	93.4	833	11	319	60
	95.7[(d)]	63.4				60
Delayed ENTOMB	301.4[(e)]	102.0	720	11	311	130
	121.9[(f)]	63.6				130

(a) Values as developed in NUREG/CR-5884. All costs include a 25% contingency.

(b) Includes costs for 121.4 years of post-closure monitoring and surveillance at original security and insurance rates.

(c) Includes costs for 121.4 years of post-closure monitoring and surveillance at reduced security and insurance rates.

(d) Includes costs for 51.8 years of pre-closure monitoring and surveillance at reduced security and insurance rates.

(e) Includes costs for 112.1 years of pre-closure monitoring and surveillance at original security and insurance rates, plus 9.7 years of post-closure monitoring and surveillance at original security and insurance rates.

(f) Includes costs for 112.1 years of pre-closure monitoring and surveillance at reduced security and insurance rates, plus 9.7 years of post-closure monitoring and surveillance at reduced security and insurance rates.

The costs for those scenarios that involve extended periods of control, during which nuclear insurance and site security costs continue, are very sensitive to the assumed annual rates for insurance and security. Thus, it is important to the viability of entombment that those insurance and security costs be minimized.

Because the costs for extended control of the entombment site and enclosure are distributed over a long time period, the present value of those costs become very small and the present-value costs of the entombment scenarios are much smaller than for the DECON alternative and less than the SAFSTOR1 alternative if that alternative were continued to 130 years after reactor shutdown, especially if the reductions in nuclear insurance and security costs can be realized.

The occupational radiation doses for the Delayed Entomb scenario and the SAFSTOR1 scenario are less than half those of the Immediate Entomb and DECON scenarios, due to postponing the disassembly and packaging activities for many years after reactor shutdown, when the principal contributor to radiation dose (^{60}Co) has decayed to insignificant levels.

Conclusions and Path Forward

The conclusions of this study are presented in this section, together with a list of the actions that USNRC needs to take to facilitate the use of the entombment decommissioning alternative.

Conclusions

- Entombment appears to be a viable decommissioning alternative for nuclear power reactors in that it protects public health and safety as defined by existing USNRC requirements. The key technical factor in determining that viability is the isolation assessment to determine the rates of release of the contained radionuclides from the enclosure and the dispersal of those radionuclides throughout the plant environs, with the accompanying radiation dose to the public or to an inadvertent intruder.

- It appears likely that entombment, excluding the highly-activated reactor vessel internals, could satisfy the performance objectives of the Radiological Criteria for License Termination Rule and for LLW burial ground closure requirements. Those objectives can also be achieved safely and at a lower cost and lower occupational radiation dose than for immediate dismantlement. Direct monitoring required to verify the integrity of the entombed structure should be minimal.

- The entombment alternatives are essentially independent of LLW disposal charge rates because little or no LLW is disposed of off-site.

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- It appears that the inadvertent intruder scenario as a radiation exposure pathway can be reasonably controlled by maintaining institutional control over the site for 130 years following reactor shutdown (assuming GTCC is removed). However, this period could be shortened if some credit could be taken for engineered barriers, such as grouting of the reactor vessel and, if necessary, by grouting of the radioactive materials remaining within the biological shielding of the plant. Within the controlled and protected environment of the entombment enclosure, the structural lifetime of the grouted material should be extremely long, thus preventing any inadvertent human access to the radioactivity present within the biological shields and the reactor pressure vessel. However, intrusion by determined and well-equipped persons can not be ruled out without continued site surveillance and protection.

- The USNRC could consider permitting in situ (entombment) disposal of the GTCC materials, assuming the results of the site and enclosure isolation assessment are satisfactory. Allowing the retention of the GTCC material within the entombment enclosure would require development of rule requirements specific to this issue, and analysis would be needed to more fully develop the technical bases.

Path Forward

Assuming that USNRC chooses to move forward with establishing a regulatory framework for implementing entombment of retired power reactors as an acceptable decommissioning alternative, there are several things that will need to be accomplished:

- Acceptable entombment designs and intruder scenarios need to first be defined and agreed upon. An isolation assessment needs to be conducted that specifically incorporates the residual source term and site characteristics relevant to nuclear power reactors. This assessment should be done for USNRC's reference PWR and BWR sites and include sensitivity cases that evaluate variations in residual source term and site characteristics across the population of existing nuclear power reactors. The results need to be evaluated within the context of existing requirements and the need for additional requirements defined (e.g., requirements for monitoring and institutional control periods).

- Should the USNRC determine that entombment of nuclear power reactors is an acceptable alternative to DECON and SAFSTOR, a supplement to the Generic EIS on Decommissioning of Nuclear Facilities will need to be prepared. This supplement would need to consider the long-term impacts from the residual radioactivity to be left in an entombed reactor, which were not considered in the original EIS.

- USNRC should establish the bases and methodologies acceptable for the performance assessments needed to determine whether a given reactor/site was suitable for decommissioning via the entombment alternative, similar to the information contained in NUREG-1573 [USNRC, 1997]. Appropriate regulatory guides must be developed and issued, including appropriate data bases containing the families of parameters and required level of conservativeness to be used in the isolation assessment analyses. For example, the amount and detail of analyses and facility characterization needed at an entombed reactor will be determined by the level of confidence required on the residual source term to satisfy the performance objectives of current USNRC rules. Satisfactory completion of the isolation assessment would be a prerequisite for USNRC's acceptance of the entombment alternative for a given reactor facility.

- Eliminating the need for full-time onsite security and surveillance staff would greatly reduce the ongoing costs associated with an extended safe storage period. Security requirements for a shutdown reactor facility which doesn't contain any special nuclear materials (i.e., irradiated or unirradiated fuel) should be clearly defined to permit electronic surveillance and monitoring from remote stations, without any personnel permanently on the site. An appropriate regulatory guide should be developed that defines the minimum acceptable monitoring and surveillance system and methods for a shutdown reactor in safe storage or entombed.

- If entombment is determined to be a an acceptable decommissioning alternative, USNRC will have to decide whether to permit entombment of GTCC material, or to explicitly exclude GTCC material from entombment enclosures.

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Appendix A

Entombment Experience in the U.S.

Richard I. Smith, P.E.

In the more than 50 cases of reactor decommissioning in the United States since 1954, only three installations utilized the onsite disposal strategy. Those three reactors were relatively small demonstration plants built by the U.S. Atomic Energy Commission in the early days of nuclear power development, namely:

- The Hallam Nuclear Power Facility,
- The Piqua Nuclear Power Facility, and
- The Boiling Nuclear Superheater Power Station (BONUS).

The unconditional release levels specified at the time of the disposal of the three demonstration power reactors were [Wheelock, 1969]:

- the external radiation hazards are "safe" if the surface dose rate from each component is less than 0.2 mrem/hr.
- the internal radiation hazards are "safe" if:
  - specific activity and solubility of the residual radioactive materials are such that the applicable non-occupational maximum permissible concentrations cannot be exceeded;
  - the total activity, times the fraction deposited upon ingestion or inhalation, is less than a non-occupational maximum permissible body burden; and
  - replacement of the total amount of the element in the standard man by the radioactive isotope of that element taken from the reactor did not exceed the allowable non-occupational body burden. This criteria is not applicable for elements such as europium for which the intestine or lung is the critical organ.

In addition, eight former plutonium production reactors owned by the USDOE that are located on the Hanford Site in the State of Washington have been considered for onsite disposal (partial dismantlement and subsequent entombment). Each of these four cases is described briefly in subsequent subsections.

The Hallam Nuclear Power Facility was a demonstration plant, located in Hallam, Nebraska. The station was initially operational in 1963, and was finally closed in 1966, with onsite disposal (in situ option) completed in 1969. The reactor was graphite-moderated, cooled using liquid sodium, and was designed to produce 256 $MW_{thermal}$. Problems with leakage of sodium into the graphite moderator and the anticipated costs for repair led to the early closure of the plant. All irradiated nuclear fuel and all bulk sodium was removed from the plant, with the residual sodium rendered inert. All residual radioactive sodium was transported to a federal nuclear installation for storage and eventual disposal. Heat exchangers and other system components were dismantled and removed. Remaining radioactive components and materials were sealed in the underground vaults of the plant. All penetrations were seal-welded, the reactor was sealed beneath two plates of 0.5 in. steel which were welded in place, and the whole entombment structure was covered with plastic film, tar, and earth. The minimum design life objective of the various seals, supplementary closures, and weatherproofing measures installed during the closure of the facility was 100 years. The details of the onsite disposal activities are reported by Atomics International [A.I.,1970]. An estimated 300,000 Ci (~ 1.1×10^4 TBq) of residual radioactivity were contained within the enclosure at the time of on site disposal. The State of Nebraska periodically inspects the site for structural integrity and radioactivity containment. Each year, gamma dose rates above the entombed structure are surveyed, and water samples are collected from six observation wells surrounding the entombment structure by the U.S. Geological Survey and are analyzed by the Nebraska Department of Health. Analytical results from the groundwater samples and the dose rate surveys have not indicated the presence of any radioactive contamination, suggesting that the containment structure is

intact and fulfilling its function [NDRH, 1995].

The Piqua Nuclear Power Facility was a demonstration plant, located in Piqua, Ohio. The station was initially operational in 1963 and was finally closed in 1966, with onsite disposal (in situ option) completed in 1969. The reactor was organically cooled and moderated, and was designed to produce 45 $MW_{thermal}$. Problems with the organic cooling system led to the early plant closure. The irradiated nuclear fuel, selected reactor core components and other radioactive materials were removed to a federal nuclear installation. The organic coolant and moderator was disposed of by burning. The reactor vessel, thermal shield, grid plates, and support barrels remained in place. The vessel penetrations were seal-welded, the vessel was filled with sand, and the enclosure penetrations were plugged. The enclosure was sealed with a waterproof barrier and a concrete cover. Contaminated piping and equipment was either decontaminated or removed from the reactor building, which was converted into a warehouse. Details of the onsite disposal activities are reported in Wheelock [1969, 1970]. An estimated 260,000 Ci (~9.6 x 10^3 Tbq) were sealed within the enclosure at the time of onsite disposal. Analyses predicted that the contained radioactivity would decay to unrestricted release levels after about 120 years. Annual environmental radiological surveys of the containment and auxiliary buildings have been performed since the closure of the entombment structure. No significant changes within the facility have been detected and no releases to the environment have been recorded [Kirsch 1989]. Nearly all readings have been less than the minimum detectable activity for the survey instruments [Millenium, 1997].

The Boiling Nuclear Superheater Power Station (BONUS) was a demonstration plant, located in Rincon, Puerto Rico. The station was initially operational in 1964 and was finally closed in 1967, with onsite disposal (in situ option) completed in 1970. The areas external to the entombment enclosure have been designated as a museum and are accessible to the public. The reactor was cooled and moderated using boiling light water, and was designed to produce 50 $MW_{thermal}$. Difficulties with the superheat system led to the early plant closure. The irradiated nuclear fuel, selected radioactive materials, and unirradiated nuclear fuel were removed to a federal nuclear installation. The penetrations through the lower portion of the reactor building were plugged and sealed, including a concrete slab which sealed off the upper surface of the engineered barrier enclosure (i.e., the reactor pedestal and shield structure). Details of the onsite disposal activities are reported in PRWRA [1970]. An estimated 50,000 Ci (~ 1.9 x 10^3 TBq) of radioactivity were sealed within the engineered barrier structure at the time of closure, comprised of about 71% ^{55}Fe, 29% ^{60}Co, and <1% ^{63}Ni. The allowable dose rate at 1 cm from the enclosure surface was required to be <0.2 mR/hr on the average, with hot spots not exceeding 1 mR/hr. A hazards analysis for the engineered barrier structure, assuming a severe earthquake followed by a tsunami, concluded that such an accident would not result in unacceptable radiation doses. Environmental radiological surveys of the site since entombment have shown no significant changes inside the containment and the entombment structure shows no indications of deterioration [Irizarry, 1993]. However, the basement of the entombed structure was flooded in 1993 due to the deterioration of an exhaust fan, which allowed ingress of rain water. A survey following this incident showed a few locations of surface contamination that potentially exceeded guideline values [Berger, 1995].

Hanford Plutonium Production Reactors The onsite disposal strategy has been selected for the eight graphite-moderated, water-cooled reactors that were constructed at the Hanford Site between 1943 and 1955 and were operated between 1944 and 1971 (from 14 to 24 years each). After shutdown, each reactor was placed in safe storage and monitored under a maintenance and surveillance program. The USDOE prepared an EIS on the decommissioning of these reactors [USDOE, 1989] which contained analyses of several decommissioning strategies for the reactors, including no action, immediate one-piece removal (onsite transfer and disposal), safe storage followed by deferred one-piece removal (onsite transfer and disposal), safe storage followed by deferred dismantling, and in situ disposal. Because the environmental impacts of these strategies did not offer a strong basis for selection among the strategies, the USDOE selected safe storage followed by deferred one-piece removal as the preferred decommissioning option for the Hanford reactors [USDOE, 1992], based on its review of environmental impacts, total project costs, and the results of the public hearing process. The preferred option is one type of onsite disposal as it consists of the decontamination and demolition of the peripheral support structures for the reactors and

transport of each reactor block intact from its present location to another area (200-West) of the Hanford site for disposal in a shallow-land disposal facility, leaving the original site suitable for release. The total residual inventory for all eight reactors was estimated in 1985 to be about 220,000 Ci. Of the eight reactors, the KE Reactor has the highest residual inventory of about 60,000 Ci, comprised of 50% ^{3}H, 31% ^{60}Co, 12% ^{14}C, 5% ^{63}Ni, and the remaining 2% being ^{41}Ca, ^{59}Ni, ^{36}Cl, ^{90}Sr, ^{93}Zr, ^{93}Mo, ^{94}Nb, ^{99}Tc, ^{108}Ag, ^{137}Cs, $^{152/154}Eu$, ^{239}Pu, and ^{241}Am [USDOE, 1989].

Presently, 105-C Reactor, the first of the eight reactors to be decommissioned, is being placed into long-term safe storage for up to 75 years. This safe storage project includes decontaminating and demolishing the surrounding support facility structures, reducing the immediately adjacent building walls to the level of the top of the reactor shield, closing off all penetrations into the shield wall, placing a new long-life stainless steel protective roof over the remaining structure, and securing the building access points. The footprint of the facility will have been reduced from about 5,000 m^2 to about 1,500 m^2. Monitoring of groundwater beneath the reactor will continue, continuous remote surveillance of the enclosure will be implemented, the interior of the enclosure will be inspected about every five years, and no maintenance is anticipated to be necessary during the safe storage period. Similar actions will be taken for the remaining seven reactors, with the possible exception of B-Reactor which has been designated a national historical engineering monument.

The in situ disposal option examined for these reactors assumed that the peripheral support structures were decontaminated and demolished, the reactor block and its shields was sealed within an engineered barrier enclosure, and the enclosure was covered with a protective earthen mound. Surfaces within the facility that are potentially contaminated would be coated with a fixative to ensure retention of contamination during subsequent activities. The major voids beneath and around the reactor block would be filled with grout or gravel as a further sealant and to prevent subsidence of the final overburden. Piping and other channels of access into the reactor building would be backfilled with grout or similar material to ensure isolation of the reactor from the surrounding environment. Finally, the reactor block enclosure and the spent fuel storage basin, together with the contained radioactive material, gravel and grout, would be covered to a depth of at least 5 m with a mound containing earth and gravel. Rip-rap on the sides of the mounds would ensure long-term structural stability and provide protection against erosion in case of a flood.

The results of the analyses for the EIS are presented in Table A-1, and the following conclusions were drawn from the EIS:

- the environmental impacts did not offer a strong basis for differentiation and selection among the alternatives,
- the consequences to the offsite population are negligible compared with the consequences from naturally occurring radiation sources,
- individual doses to persons who drill wells near the waste sites and who use the water for drinking and/or irrigation of small family farms can exceed existing drinking-water standards, and
- the scenarios involving contact with or intrusion into the waste form or entombed structure indicate significant adverse consequences to those who ignore warnings and intrude into the wastes.

Additional entombment actions are being considered for the final disposition of the high-level liquid waste tanks at the Hanford site [USDOE, 1987] (and possibly the Savannah River site), after the liquids and sludges have been removed. One option anticipated that any residues of the wastes remaining in the tanks following the removal efforts would be grouted, the tanks backfilled with sand and gravel to prevent eventual tank dome subsidence, and seal the openings. The entombed tanks would be within

controlled areas on federal reservations, and thus would be under long-term institutional control

Because a number of years will elapse before these actions take place, the final decision on the actual closure method will not be made until after the wastes have been removed.

Table A-1. EIS Results for the Eight Surplus Production Reactors at the Hanford Site
[USDOE, 1989]

Alternative	Total Cost (millions of 1986 $)	Occupational Radiation Dose (person-rem)	Drinking Water from		Full-Garden Water Well		10,000-yr Integrated Population Whole-Body Dose from River Pathway (person-rem)	Maximum Intruder Whole-Body Dose (rem/yr)
			Maximum Whole-Body Dose (mrem/yr)	Time (years after disposal)	Maximum Whole-Body Dose (mrem/yr)	Time (years after disposal)		
No Action	41	24	1,200	140	35,000	140	50,000	10
Immediate One-Piece Removal	191	159	40	6,090	1,400	6,160	1,900	20
Safe Storage Followed by Deferred One-Piece Removal	198	51	40	6,090	1,400	6,160	1,900	20
Safe Storage Followed by Deferred Dismantlement	217	532	40	6,090	1,400	6,160	1,900	20
In-Situ Decommissioning	181	33	30	1,120	460	1,120	4,700	10

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Appendix B

Isolation Assessment

Steven M. Short, P.E.

The key evaluation in determining the viability of entombment is the assessment of the isolation of the enclosure in retaining the residual radioactivity and the analysis of the doses to the public arising from dispersal of the contained radioactivity throughout the environs surrounding the enclosure over time. Factors important to the enclosure performance assessment include:

- Radioactivity inventory present within the entombment enclosure
- Long-term physical integrity of the entombment enclosure structure
- Physical and chemical forms and the adsorption and solubility of the individual radioactive species in various qualities of water
- Transport of the dissolved individual radioactive species through the entombment enclosure materials
- Transport of the released radioactive species throughout the environment outside of the entombment enclosure structure.

These factors are discussed briefly in subsequent subsections.

Radioactivity Inventory - The inventory of radioactivity present within the entombment enclosure (i.e., the types of radioactive species and the activity levels of each species, and the physical form of the material containing those radioactive species) is very important information in the determination of the viability of entombment for a given facility. Because the entombment enclosure structure cannot be assured to remain impenetrable forever, it is necessary to know which radioactive species are present in what quantities at what times. An estimate of the source term for radionuclides of concern in an entombed PWR are presented in Table B-1 [Smith, 1978].

For short-lived species such as ^{60}Co, its 5.27-year half-life assures that this species will decay to negligible levels well within the anticipated lifetime of the enclosure structure. For intermediate-lived species such as ^{137}Cs and ^{63}Ni, their 30- and 100-year half-lives suggest that containment for about 300 to 1000 years would be required for these species to decay to negligible levels. For long-lived species such as ^{94}Nb and ^{59}Ni, their 20,000-year and 80,000-year half-lives assure that these species will be present essentially forever, well beyond the possible lifetime of the enclosure structure. Thus, the rates at which these various species become available via corrosion and degradation of the matrix and disperse throughout the environment over time may be the controlling considerations in determining entombment viability.

The calculated activity levels for the principal radionuclides in the activated metals of the carbon steel reactor pressure vessel and the stainless steel vessel liner of the reference PWR Smith [1978] are shown in Table B-2, at reactor shutdown and after 130, 1,000, and 10,000 years of decay. These calculations assumed the removal of all of the vessel internals for storage and disposal, leaving only the

vessel and its liner. Thus, no GTCC materials were included in this analysis. From the table, it can be seen that, over time spans approaching the confinement lifetime of a HLW repository, the very long-lived radionuclides remain at a nearly constant level.

Forms and Solubility Rates of Radioactive Species - The rates at which the radioactive species can escape from the entombment enclosure structure are strongly dependent upon physical form, solubility, and adsorption and desorption processes of each species. For example, if the species is in the form of surface contamination that resulted from a spill of contaminated water, that material would readily go back into solution upon contact with water, and could be transported out of containment as readily as the water entered into the containment enclosure. Considering the high integrity of the reactor containment structure, it seems unlikely that any significant amount of water could penetrate into the interior of the structure in less that 500 years. Thus, the shorter-lived radionuclides commonly found in surface contamination within reactor facilities and within contaminated process systems will have essentially decayed to insignificant levels before any water could reach, dissolve, and transport the surface contamination to the surrounding environs.

Table B-1. Residual Source Term in the Reference PWR*

		At Shutdown (Curies)			60 Years After Shutdown (Curies)		
Radionuclide	Half-life (years)	Greater Than Class C	Class A, B, and C LLW	Total	Greater Than Class C	Class A, B, and C LLW	Total
Cesium-134	2.1	NA	2.70e-03	2.70e-03	NA	--	--
Iron-55	2.7	2.15e+06	1.87e+04	2.17e+06	4.39e-01	3.83e-03	4.43e-01
Cobalt-60	5.27	1.55e+06	4.54e+03	1.55e+06	5.78e+02	1.70e+00	5.80e+02
Tritium	12.3	NA	7.53e-04	7.53e-04	NA	2.56e-05	2.56e-05
European-152	12.7	NA	1.79e+01	1.79e+01	NA	6.79e-01	6.79e-01
Strontium/Yttrium-90	27.4/<0.1	NA	1.86e-05	1.86e-05	NA	4.08e-06	4.08e-06
Cesium-137	30.1	1.56e-02	5.72e-01	5.87e-01	3.92e-03	1.44e-01	1.48e-01
Nickel-63	100	2.01e+05	4.62e+02	2.02e+05	1.33e+05	3.05e+02	1.33e+05
Argon-39	269	NA	1.30e+00	1.30e+00	NA	1.12e+00	1.12e+00
Molybdenum-93	3,500	6.22e-01	2.88e-02	6.51e-01	6.15e-01	2.84e-02	6.43e-01
Carbon-14	5,750	2.50e+02	8.19e-01	2.51e+02	2.48e+02	8.13e-01	2.49e+02
Niobium-94	20,000	7.99e+00	1.01e-02	8.00e+00	7.97e+00	1.01e-02	7.98e+00
Technetium-99**	21,000	1.69e-01	1.84e-03	1.71e-01	1.69e-01	1.83e-03	1.71e-01
Nickel-59	80,000	1.33e+03	3.86e+00	1.33e+03	1.33e+03	3.86e+00	1.33e+03
Calcium-41	80,000	NA	2.30e-01	2.30e-01	NA	2.30e-01	2.30e-01
Chlorine-36	308,000	NA	2.76e-04	2.76e-04	NA	2.76e-04	2.76e-04
Potassium-40	1.26e+09	NA	1.23e-03	1.23e-03	NA	1.23e-03	1.23e-03
Total Activity		3.90e+06	2.38e+04	3.92e+06	1.35e+05	3.14e+02	1.35e+05

* Greater Than Class C includes the Shroud, Lower Core Barrel, Thermal Shields, Upper and Lower Grid Plates, and Lower Support Columns. Class A, B, and C LLW includes the Reactor Vessel Inner Liner and Wall, Other Miscellaneous Reactor Vessel Internals, Concrete Bioshield, and Corrosion Products.

* * Ratioed from Niobium-94/Technetium-99 in Oak, [1980]

*** These source term estimates assume aggressive chemical decontamination of the primary system.

On the other hand, if the radionuclides are bound into a metal matrix such as the neutron-activated carbon steel vessel and stainless steel vessel liner, then the matrix essentially has to be corroded/dissolved to free the radioactive species for transport out of the containment enclosure, and corrosion of stainless steel in ordinary water is quite slow. Some representative corrosion rates for

carbon steel and stainless steel in varying types of water and soil are presented in Table B-3 (from Appendix C of this report), together with calculated release rates of the radionuclides from the vessel and liner. As can be seen in the table, the amount of radioactivity released per year into the water or soil in which the vessel and liner are immersed is very small.

The result for the vessel in low chlorine soil appears to be anomalous, in that the corrosion rate for carbon steel is higher than in sea water. In any event, the releases from the entombed vessel and vessel liner apparently would not exceed 10 to 15 millicuries per year, and probably less.

Table B-2. Activated Radionuclides in Vessel Liner and Vessel Wall as a Function of Decay Time

		Activated Stainless Steel Vessel Liner [b]				
Radionuclide	Half-life (yr)	Curies/m^3	Ci/g (0)	Ci/g (130)	Ci/g (1,000)	Ci/g (10,000)
Cobalt-60	5.27	2.50e+03	3.11e-04	1.17e-11	8.91e-69	0.00e+00
Nickel-63	100	3.60e-02	4.48e-09	1.82e-09	4.38e-12	3.55e-39
Molybdenum-93	3,500	1.20e-03	1.49e-10	1.46e-10	1.22e-10	2.06e-11
Carbon-14	5,750	4.00e-01	4.98e-08	4.90e-08	4.41e-08	1.49e-08
Niobium-94	20,000	9.50e-03	1.18e-09	1.18e-09	1.14e-09	8.36e-10
Technetium-99[a]	21,000	2.01e-04	2.50e-11	2.49e-11	2.42e-11	1.80e-11
Nickel-59	80,000	3.00e+00	3.73e-07	3.73e-07	3.70e-07	3.42e-07
Total Activity		3.45e+00	3.11e-04	4.25e-07	4.15e-07	3.58e-07

		Activated Carbon Steel Vessel Wall [b]				
Radionuclide	Half-life (yr)	Curies/m^3	Ci/g (0)[c]	Ci/g (130)[c]	Ci/g (10^3)[c]	Ci/g (10^4)[c]
Cobalt-60	5.27	7.50e+01	9.33e-06	3.50e-13	7.12e-63	0.00e+00
Nickel-63	100	3.80e+00	4.73e-07	1.92e-07	4.62e-10	3.75e-37
Molybdenum-93	3,500	1.30e-03	1.62e-10	1.58e-10	1.33e-10	2.23e-11
Carbon-14	5,750	1.90e-02	2.36e-09	2.33e-09	2.10e-09	7.08e-10
Niobium-94	20,000	0.00e+00	0.00e+00	0.00e+00	0.00e+00	0.00e+00
Technetium-99*	21,000	0.00e+00	0.00e+00	0.00e+00	0.00e+00	0.00e+00
Nickel-59	80,000	3.20e-02	3.98e-09	3.98e-09	3.95e-09	3.65e-09
Total Activity		7.89e+01	9.81e-06	1.98e-07	6.64e-09	4.38e-09

[a] Ratioed from Niobium-94/Technetium-99 in Oak, [1980]

[b] Liner and vessel surface areas ~59 m^2 each

[c] Decay time in years

Table B-3. Release of Activated Radionuclides from Reactor Vessel via Corrosion

Water Type	Corrosion Rates ($g/m^2/yr$)		Mass Released (g/yr) [(a)]		Curies Released per year[(b)]	
	Liner	Vessel	Liner	Vessel	Liner	Vessel
low Cl soil©	4.5	1254	267	74,000	0.113×10^{-3}	14.6×10^{-3}
sea water[(d)]	5.7	555	8370	32,700	3.56×10^{-3}	6.48×10^{-3}

(a) Surface Area ~ 59 m^2 each.
(b) Activity at 130 years decay.
© See Appendix A, [Romanoff, 1957].
(d) See Appendix A, [Schumacher, 1979].

Transport through Containment - The containment enclosure of a power reactor is designed to retain liquids and gases that could be generated in a major accident. The structure is generally rather thick concrete (~ 4 ft.), contains much structural steel reinforcing rod, and usually has a sealed steel liner on the inside of the concrete. The concrete is usually very high strength (3000 to 6000 psi compressive strength). Under normal conditions, barring physical damage, the enclosure is almost impervious to water intrusion for many years.

Even without any cataclysmic events, it is probable that pathways will develop through the containment enclosure over time, in the form of fine cracks or fissures, that will permit the migration of ground or surface water into the enclosure where it can immerse the enclosure contents and over time corrode and degrade those materials. Once released, the radioactivity will be mobile and will migrate out of the containment enclosure at rates dictated by the solubility of the materials and the transmissibility of the enclosure material. It is anticipated that the release of the dissolved radioactivity will occur via a diffusion process through the very fine pores and cracks in the enclosure, because there will be no internal pressurization of the enclosure to drive the contaminated water through the enclosure wall. In a steady-state condition, the rate of release of radioactivity outside of the vessel should at worst be about equal to the rate of release from the metal matrix by corrosion, which was shown previously in Table B-3 to be quite small.

Long-Term Enclosure Integrity - There is historical evidence that concrete structures covered with soil have remained intact for several thousand years. Using today's materials and techniques, it appears that most containment structures would remain essentially intact and resistant to water intrusion for 500 years or more [Clifton, 1989; Walton, 1990]. Thus, the buried portion of the reactor containment structure should remain intact and reject water intrusion for a long time, barring some cataclysmic natural event such as an earthquake that exceeds the structural design basis or a nearby volcanic eruption. Even such an earthquake would be unlikely to create major breaches in the highly reinforced enclosure structure, but might result in some cracks sufficiently large to permit fluid flow into and out of the structure. In this situation, the release rate of radioactivity into the environment would be controlled by the dissolution rate of the radioactive materials in free-flowing water, as discussed above.

Dispersal through the Environment - Dispersal of the dissolved radioactivity once it has escaped from the entombment enclosure will depend upon the soil materials and ground and surface water volumes in contact with the enclosure and the pathways for those waters through the local soils. Some soils act as a chemical filter, extracting the dissolved contaminants from the water in much the same manner as an ion

exchange column, thereby preventing widespread dis
persal of the radioactivity throughout the environs. Other soil types tend to be more chemically neutral, permitting the dissolved contaminants to move relatively freely with the local groundwater flows. Modeling of the local groundwater pathways and flow rates for the contaminated waters released from the entombment enclosure will be an essential aspect of determining the suitability of a given site for use as an entombment disposal site. Some typical values of transport coefficients (K_d) for elements contained in the vessel and vessel liner through different types of media are shown in Table B-4. As can be seen from the table, there is a wide variability among the elements and the different media. Thus, careful attention to selection of K_d's appropriate for the element and the medium is important. A more detailed discussion of this subject is given in Appendix D.

Table B-4. Transport Coefficients for Typical Elements and Soils

Media	Element	K_d (mL/g) Literature	Hanford	Savannah River
Concrete Grout / Soil	Carbon	10^3 - 10^4 / 3 - 8	2,625 / 0	5,000 / 2.4
	Nickel	5 - 50 / ---	125 / 3	--- / ---
	Niobium	--- / ---	2 / 0.67	--- / ---
	Molybdenum	--- / ---	--- / ---	--- / ---
	Technetium	1 - 100 / 2 - 5	2 / 0	700 / 0.36

Results of Analog Performance Assessments

Since no isolation assessments have been performed on any of the nuclear reactors entombed in the U.S. to date, it is not possible to provide any isolation assessment results for reactor entombment scenarios similar to those being considered by the USNRC. The entombment period and final unrestricted release of the USDOE reactors discussed in Appendix A was based strictly upon decay of the radionuclides of concern and the fact that institutional controls would be maintained for the entire entombment period (determined to be 100 to 120 years before unrestricted release was possible). However, a number of performance assessments have been performed for low-level waste (LLW) disposal sites, which can be used as analogs for the entombed reactor structures being considered in this study. While there are major differences between the two (e.g., site characteristics, source term, and engineered barriers), the results of performance assessments for LLW burial sites can provide valuable insights as to what the performance criteria would be for entombed nuclear power reactors.

This analysis summarizes the results of performance assessments for three different facilities located at two different sites:

1) Burial Ground 218-E-12B located on the Hanford Site in southeast Washington. This burial ground is being used for the disposal of up to 120 decommissioned naval submarine reactor compartments. The reactor vessels are used as the burial container and are filled with low-density grout to provide additional containment of the activated metals.

2) Grout Disposal Facility located on the Hanford Site in southeast Washington. This facility was to be used for the disposal of approximately 120 million liters of grouted liquid LLW, solidified into a grout monolith. The facility was to be composed of about 33 reinforced concrete vaults (each having dimensions of 37.6-m long, 15.4-m wide, and 10.4-m deep) filled with the grouted liquid LLW.

The vault design includes multiple engineered barriers to isolate the grouted wastes from the environment and warning marker systems to deter inadvertent intrusion.

3) Saltstone Disposal Facility located on the Savannah River Site in South Carolina. This facility is being used for the disposal of up to 730 million liters of grouted liquid LLW, solidified into a grout monolith. The facility is to be composed of up to 15 reinforced concrete vaults (each having dimensions of 180-m long, 60-m wide, and 7.6-m deep) filled with the grouted liquid LLW. The vault design includes multiple engineered barriers to isolate the grouted wastes from the environment.

The design concepts and grout formulations for each of these facilities evolved over time, based on the results of preliminary performance assessments, to the point where each has now been shown to meet USDOE's performance objectives for the protection of the general public, inadvertent intruder, and groundwater resource. These performance objectives are provided in Table B-5, along with a comparison to the actual results for the performance assessments conducted for the three facilities.

While there are many site-specific environmental and engineered barrier characteristics needed to conduct a performance assessment, experience has shown that radionuclide release, and ultimately dose estimates, are primarily influenced by the following three factors: 1) the chemical and physical properties of the radionuclides, primarily the retardation characteristics (K_d) of the soils and engineered systems, 2) the physical properties and integrity of the waste form and engineered containment, primarily the rate of degradation (including corrosion) of the engineered barrier, and 3) the recharge rate of water entering the engineered system. Other sections of this report have provided a focussed discussion of the first two of these items relative to values for these parameters that would be expected in a performance assessment of an entombed nuclear power reactor. The last item, recharge rate, is highly site-specific but would generally range from less than 0.1 cm/year at a dry, arid site such as Hanford (and where the engineered barrier system is assumed to be effective at diverting recharge away from the waste site for up to 1000 years) to greater than 40 cm/year at a wet, humid site such as Savannah River (and where the engineered barrier system is assumed to not be effective at diverting recharge away from the waste site).

Values for other site and engineered system characteristics assumed in each of the three assessments are presented in Table B-6. Also provided are the source terms assumed for each of the major radionuclides. The retardation characteristics (K_d) of the soils and engineered systems are given in Appendix D

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Table B-5. DOE Performance Objectives for LLW Disposal (DOE, 1988) and Comparison to Performance Assessment Results

Performance Objective Category	Exposure Path/Scenario	Performance Objective	Time Period	Performance Assessment Results		
				Naval Reactors[a]	Grout Disposal Facility[b]	Saltstone Disposal Facility[c]
General Public Protection	Farm Family	25 mrem/yr	10,000 years	0.00097 mrem/yr	5.2 mrem/yr	50 - 110
	Residential Family	25 mrem/yr	10,000 years	Not Evaluated	8.2 mrem/yr	10 - 70
	Population - River	500 person-rem/yr	10,000 years	<0.00053 person-rem/yr	<0.1 person-rem/yr	Not Evaluated
	Population - Groundwater	500 person-rem/yr	10,000 years	Not Evaluated	2.8 person-rem/yr	Not Evaluated
Inadvertent Intruder Protection	Drilling - Acute	500 mrem	@ 500 years	Not Evaluated	8.4 mrem/yr	Not Evaluated
	Post-Drilling Habitation (Chronic)	100 mrem/yr	@ 500 years	Not Evaluated	72 mrem/yr	10 mrem/yr
Groundwater Resource Protection	Groundwater	4 mrem/yr	10,000 years	0.0003 mrem/yr	0.4 to 6.9 mrem/yr	0.03 mrem/yr

a. [Rhoads, 1994].
b. [Kincaid, 1995].
c. [WSRC, 1992].

Table B-6. Comparison of Parameters Used in Each of the Three Performance Assessments

Category	Radionuclide	Naval Reactors[a]	Grout Disposal Facility[b]	Saltstone Disposal Facility[c]
Source Term (Ci)	^{3}H	---	619	19,000
	^{14}C	---	244	6.5
	^{59}Ni, ^{63}Ni	26,000	14,500	20.2
	^{79}Se	---	366	320
	^{90}Sr	---	1,570,000	680
	^{94}Nb	---	0.159	---
	^{99}Tc	---	6,310	65,000
	^{126}Sn	---	1,110	130
	^{129}I	---	14.6	20
	^{135}Cs, ^{137}Cs	---	16,100,000	20,000
	^{152}Eu, ^{154}Eu, ^{155}Eu	---	39,100	980
	^{237}Np	---	53.6	.058
	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu	---	1,400	82
	^{241}Am	---	2,840	130
	^{242}Cm, ^{243}Cm, ^{244}Cm	---	11.5	0.7
Site Characteristics	Distance to River	9.6 km	9.6 km	15 km
	Depth to Groundwater	61m	61 - 70 mt	7 - 18 m
	Recharge Rate	0.1 - 6.0 cm/yr	0.1 - 5.0 cm/yr	2 - 40 cm/yr
	K_d	See Table C-1, Appendix C		
	Soil Type	gravel/sand/silt	gravel/sand/silt	sandy/loamy/clayey
Engineered System Parameters	Concrete and/or Grout Degradation Rate	---	through-wall cracking assumptions	100 mm/1000 yrs
	Stainless Steel/Inconel Corrosion Rate	0.02/0.01 mg/dm^2-yr	---	---
Computer Models	Vadose Zone Transport	TRANSS	PORFLOW	PORFLOW-3D
	Unconfined Aquifer Transport	CFEST	SLAEMS	PORFLOW-3D
	Dose	GENII	GENII	Hand Calculations

a. [Rhoads, 1994]. b. [Kincaid, 1995]. c. [WSRC, 1992].

Appendix C

Corrosion Issues Related to Entombment of Nuclear Reactors

Michael J. Danielson

Background

Once exposed, conventional transport processes, such as ground water mobility, can operate to carry these materials to the general populous. Consequently, corrosion processes define when the source term for the input of radioactive elements into the external environment starts as well as its magnitude.

Radioactive elements are present in various reactor components because (1) of direct transmutation of certain elements within the metal into radioactive elements, and (2) transmutation into a radioactive element which is soluble in the cooling water or other medium and which is later transported to another location and incorporated into an oxide film, porous layer, or other coating. Both of these processes result in a gradient of radioactive material such that the highest concentrations are located closest to the nuclear source or the cooling water. The entombment process would result in the sealing or closing of all the metallic pressure boundary components. The greatest radioactive source term input would take place when there was complete penetration of the pressure boundary components, and predictions for this period of time would be valuable in evaluating the propensity for leaks of radioactive materials into the external environment. It should be pointed out that uniform corrosive attack will gradually release some radioactive elements that are incorporated throughout the wall of the component.

Literature

The three major corrosion degradation processes for metals are stress corrosion cracking, pitting, and uniform corrosive attack. Stress corrosion cracking which leads to fracture of the component can probably be discounted because the major tensile stresses are largely gone, once the components are taken from service. Pitting and uniform corrosive attack are the most likely processes with pitting being the more aggressive process. Also, these forms of attack result in the formation of soluble corrosion products, which may lead to a direct entry of the radioactive elements into solution as the base metal is corroded. The corrosive environment is extremely important in defining the rate of the corrosive processes, but it cannot be clearly defined since there are a variety of reactor locations (and soils associated with each environment). As a general rule, acidic environments are worst than neutral or alkaline pH environments for increasing the rate of general attack but have little or no effect on pitting. Chlorides in the environment increase the rate for all forms of attack but are particularly important to increasing pit penetration rates. Microbiologically Influenced Corrosion (MIC) can also be very important since the organisms can create a localized environment rich in such aggressive components as hydrogen sulfide and acid chlorides which greatly increase the rate of pitting. The signs of MIC are so ubiquituous that it is difficult to clearly show a well-defined influence for the micro-organisms.

For this paper, the rate of the corrosion process will be bracketed using a natural ground water environments using the data of Romanoff [1957] and seawater environments. This database will naturally encompass the effects of MIC though its role cannot be clearly defined. There are a variety of materials used in nuclear reactors, but for this analysis only two classes of material will be examined--plain carbon steels and stainless steels (such as 304).

Stainless Steel Results

Stainless steels are considerably more prone to pitting attack than general attack. The corrosion results are shown in Table C-1 for several ground water environments. Docas clay at Cholame, CA is the environment highest in chloride from the Romanoff study but it is considerably lower in annual precipitation than the Lake Charles clay at El Vista, TX, and this difference would probably act to reduce the pitting rate at Cholame, CA. For comparison, the data from soil #62 was included in which no pitting was detected. There is no apparent reason why one environment is better than another. Using the pitting rate found in the low chloride soil at El Vista, TX, 2.3 inches of stainless steel would be penetrated in a thousand years of exposure.

Table C-1. Corrosion Results for 304 Stainless Steel [Romanoff, 1957]

Soil	Environment	Pitting Rate (max depth)	Uniform Attack Rate
#56, Lake Charles Clay; El Vista, TX. Very poor aeration	pH = 7.1, Low Cl Mean Rainfall = 49 in/y Mean Temp. = 69 °F	max depth 32+ mils in 14 years (2.3 mils/y)	0.20 oz/ft^2 (61 g/m^2) in 14 years (0.022 mils/y)
#64, Docas Clay; Cholame, CA. Fair aeration	pH = 7.5, high Cl (288 ppm Cl) Mean Rainfall = 16 in/y Mean Temp. = 58 °F	max depth 7 mils in 14 years (0.5 mils/y)	0.04 oz/ft^2 (12 g/m^2) in 14 years (0.0043 mils/y)
#62, Susquehanna Clay, Meridian, MS	pH = 4.5 Mean Rainfall = 53 in/y Mean Temp. = 64 °F	no pitting in 14 years (one of two specimens)	0.0004 oz/ft^2 (0.12 g/m^2) in 14 years (4.3E-5 mils/y)

+ means complete penetration of specimen

300 series stainless steels are considered unsuitable for seawater applications unless they are subjected to high velocity flow rates or cathodic protection. Some corrosion data for a 302 stainless (nominally the same as a 304 stainless steel) in seawater is shown in Table C-2. These materials were exposed to 8 years of testing at Panama City.

Table C-2. Corrosion Results for 302 Stainless Steel [Schumacher, 1979]

Environment	Pitting Rate (Max Depth)	Uniform Attack Rate
immersed in seawater, Panama City	max depth 261 mils in 1 year	average 0.69 mils/y over 8 years

Carbon Steel Results

The results for a low carbon steel (open hearth) are shown in Table C-3 for the same soil environments as the stainless steel exposures. The Romanoff study incorporating this steel started in 1937 and used a steel that was representative of good steel-making practice for that time but which does not duplicate pressure vessel and piping steels of the present era. Using the pitting rate found in the low chloride soil at El Vista, TX, 250 inches of carbon steel would penetrated in 1000 years.

Table C-3. Corrosion Results for Carbon Steel (0.06% C, 0.098% Mn, 0.029% S, 0.069% P, 0.54% Cu, 0.13% Mo) [Romanoff 1957]

Soil	Environment	Pitting Rate (Max Depth)	Uniform Attack Rate
#56, Lake Charles Clay; El Vista, TX. Very poor aeration	pH = 7.1, Low Cl Mean Rainfall = 49 in/y Mean Temp. = 69 °F	max depth 250+ mils in 12.7 y (average 19.7+ mils/y)	45.2 oz/ft^2 (13,700 g/m^2) in 11.1 years (average 6.1 mils/y)
#64, Docas Clay; Cholame, CA. Fair aeration	pH = 7.5, high Cl (288 ppm Cl) Mean Rainfall = 16 in/y Mean Temp. = 58 °F	max depth 129 mils in 12.7 y (average 10.2 mils/y)	14.7 oz/ft^2 (4,480 g/m^2) in 11.1 years (average 2.0 mils/y)

+ means complete penetration of specimen

The Romanoff study suffers from not including seawater or other high chloride environments in the test matrix. Long-term (16 year) exposure data for a carbon steel in tropical seawater is reported in Table C-4 [Schumacher, 1979].

Table C-4. Corrosion Results for Carbon Steel in Tropical Seawater [Schumacher, 1979]

Environment	Pitting Rate (Max Depth)	Uniform Attack Rate
immersed in seawater, Panama City	max depth 155 mils in 16 years (average 9.7 mils/y)	average 2.7 mils/y over 16 years

Other seawater data from test sites located worldwide indicate an insensitivity to site location and give an *average* uniform corrosion rate of 3.4 to 4 mils/y over the five year study. This same study [Schumacher, 1979] pointed out that the *maximum* pit penetration rate is up to eight times greater than the *average* uniform penetration rate calculated from weight loss. Low alloy, high strength steels, such as used in reactor pressure vessels, corrode at a rate that can be 1/5 that of plain, carbon steels in marine environments. All the corrosion results reported in this paper are for plain, carbon steels; consequently, using these numbers in release calculations should lead to conservative estimates for the time when the vessel will be breeched or for corrosion-controlled release rates.

Conclusions

The corrosion rate data for stainless steels and plain carbon steels in the soil and seawater environments considered here indicate that significant penetration rates are possible. Pit penetration rates are considerably higher than general attack rates for both classes of material. Pit penetration rates can be similar for both stainless steels and carbon steels, depending upon the environment, though the uniform attack rate is much lower for stainless steels.

The original Romanoff study does not consider soils that are very high in chloride. This study is still ongoing and there should be results from longer term exposures (1922 burial is the earliest work). Periodically, samples are removed and new materials are added as metallurgical progress creates more modern alloys or alloys that have benefited from advances in processing.

Ultimately, it is the environment which controls the corrosion rate, and that environment is highly variable. This white paper has examined only a few environments in its attempt to estimate the corrosion rates for stainless and carbon steels. In this cursory examination, references reporting even higher corrosion rates may have been overlooked.

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Appendix D

The K_d Adsorption/Desorption Model

Steven M. Short, P.E.

Introduction

The most commonly accepted nuclear waste site performance assessment methodology for modeling the partitioning of radionuclides between the aqueous solution and solid surfaces after they have leached from the initial waste forms is the empirical K_d approach. This approach is generally chosen over other, more mechanistic, approaches that provide fundamental data for understanding the physicochemical processes that are actually controlling the radionuclide retardation because of 1) the time required to complete mechanistic studies, 2) the results of such studies are not as readily useable in performance assessments, and 3) more sophisticated transport codes are needed in such studies. The relatively simple K_d retardation model is technically defensible for use in performance assessments when 1) the groundwater and adsorbents along the flow path are homogeneous (constant in composition) and not changing in time and 2) K_d's are used for bounding calculations. Generally, the scenarios evaluated in performance assessments are considered bounding or worst-case situations. In these situations, it is defensible to select a constant K_d value that is representative of conservative values found on adsorbents from the geologic setting. Typically, a K_d value that represents a minimum value observed in experiments is chosen for worst case or conservative analyses.

Discussion

The distribution coefficient, K_d, is a value which refers to the ratio of mass or activity of a radionuclide present in the solid phase to the mass or activity present in solution. Radionuclide adsorption is influenced by many attributes of the engineered system, waste form, and geologic environment. For this reason, values for K_d's used in performance assessments are typically determined from laboratory measurements of site-specific materials and media. However, the measurement of K_d values that accurately represent the adsorption processes of radionuclides is not a trivial task. There are studies reported in the literature where several laboratories were provided samples of the same rock and asked to determine K_d values for specific radionuclides [Ames, 1978 and 1991 NEA Sorption Workshop]. The results of the experiments yielded K_d values for each radionuclide that spanned a range of 3-to-4 orders of magnitude. The variability in these studies was attributed to several factors, including mixing rate, incomplete removal of the fine particulate matter, the method of radionuclide addition to the solution, adsorption onto the walls of the container or the filtration media, the solution-to-rock ratio, non-linear absorption isotherms, and temperature. Therefore, careful attention must be given to ensure that the values determined in the laboratory are obtained under conditions that mimic, as closely as possible, those expected in the field.

While it has been stressed that K_d values are site-specific and should be determined for each individual application, it is of interest to compare the values of K_d's used in accepted performance assessments of sites located in radically different environments with measured K_d's reported in the literature. Table D-1 provides this comparison for several elements that would be of particular concern in the entombment of nuclear power reactors. The two sites chosen for comparison were DOE's Hanford Site located in southeast Washington (desert, arid environment having sandy/silty soil) and DOE's Savannah River Site located in South Carolina (humid environment having loamy/clayey soil). The K_d values for the two DOE sites were used in performance assessments for the disposal of cemented liquid

low-level radioactive wastes in reinforced concrete vaults.

Conclusions

As can be seen in Table D-1, vastly different values for K_d may be applicable from one site to the next. There are any of a number of site geochemistry conditions that can cause this, including 1) the expected speciation of the radionuclides under the various conditions being modeled, 2) solubility limits for each of the radionuclides under the various conditions being modeled, and 3) the potential mobilization of radionuclides resulting from the presence of chelating agents. While these will not be discussed in detail here, the environmental conditions of an entombed nuclear power reactor are expected to be as follows: 1) the equilibrium solubility limits of the radionuclides of concern are not expected to be exceeded, therefore precipitation of released radionuclides will not provide an additional mode for the retardation of its transport (although this potential retardation effect should be considered in any evaluation given that water within the containment is expected to have a high pH, >10, that will act to substantially reduce equilibrium solubility limits) and 2) the residual presence of man-made organic chelating agents (such as EDTA, oxalic acid, and citric acid) used in the decontamination of equipment and systems will act to mobilize cationic radionuclides (such as Nickel).

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Table D-1. Comparison of Selected Site-Specific K_d Values to Literature K_d Values

Media	Element	K_d (mL/g)		
		Literature[a,b,c,d,e]	Hanford[f]	Savannah River[g]
Concrete/Grout	C	10^3 - 10^4	2,625	5,000
	Nickel	5 - 50	125	---
	Sr	2 - 5	125	10
	Nb	---	2	---
	Mo	---	---	---
	Tc	1 - 100	2	700
	I	1 - 30	0	30
	Cs	1 - 5	125	2
	Pu	10^3 - 10^4	2,625	5,000
	Am	10^3 - 10^4	2,625	5,000
Soil	C	3 - 8	0	2.4
	Ni	---	3	---
	Sr	5 - 10	3	10
	Nb	---	0.67	---
	Mo	---	---	---
	Tc	2 - 5	0	0.36
	I	0.1 - 0.8	0	0.6
	Cs	30 - 300	3	100
	Pu	---	21	100
	Am	---	21	150

a. [Allard, 1978]
b. [Ewart, 1988]
c. [Nacarrow, 1988]
d. [Allard, 1984]
e. [Conca, 1990]
f. [Kincaid, 1995]
g. [WSRC, 1992]