United States Nuclear Waste Technical Review Board

Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel

December 2010

This report was prepared for the U.S. Nuclear Waste Technical Review Board by Dr. Douglas B. Rigby, staff member, in support of the Board's analysis of issues associated with extended dry storage and transportation. The effort is led by Dr. Andrew C. Kadak, Board member.

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Introduction

The U.S. Nuclear Waste Technical Review Board (Board) is tasked by the amendments to the Nuclear Waste Policy Act of 1982 to independently evaluate U. S. Department of Energy (DOE) technical activities for managing and disposing of used nuclear fuel and high-level radioactive waste. This report was prepared to inform DOE and Congress about the current state of the technical basis for extended dry storage[1] of used fuel and its transportation following storage. The Board expects that the report also will be valuable in informing the Blue Ribbon Commission on America's Nuclear Future, and other interested parties, on these issues.

When the used nuclear fuel that is currently stored at commercial nuclear power plant sites will be transported to other locations is not known. Understanding the length of time that used fuel can be stored, without the fuel or the storage system components degrading to the extent that the ability to meet the regulatory requirements for continued storage is affected, is a primary concern. In addition, understanding how the condition of the used fuel changes with time is important to determining when this may affect the ability to transport the fuel without significant risk of damage or release of radioactive materials. Finally, being able to predict confidently how used fuel will behave when it is handled after transportation to a repository or a processing facility also is necessary.

This report presents the results of a review of publicly available literature and published information on research completed to date related to extended storage and transportation of used fuel. The Board reports these results without challenging the technical findings of researchers but believes that they form a suitable basis for the evaluation presented here. In addition, regulatory authority, National Laboratory, and industry experts have been consulted to confirm the current state of knowledge and the research and development recommendations to enhance confidence in the evaluation of extended storage included in this report.

Background

Figure 1 shows graphics of typical pressurized- and boiling-water reactor fuel assemblies consisting of fuel rods arranged in assemblies. After irradiation, these constitute the "used fuel" that is stored at reactor sites in pools and in dry-storage systems.

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[1] U.S. nuclear utilities are operating dry storage facilities for used fuel that are licensed for operating periods of up to 60 years. The fuel in these facilities and the used fuel that will be discharged in the foreseeable future may need to remain in storage for much longer periods. Some have suggested that this period could extend as long as 300 years. This report evaluates the technical basis for dry storage of used fuel during such extended periods and does not encompass extended wet storage of fuel. In this report, the term "fuel" refers to both the uranium pellets and the metal cladding.

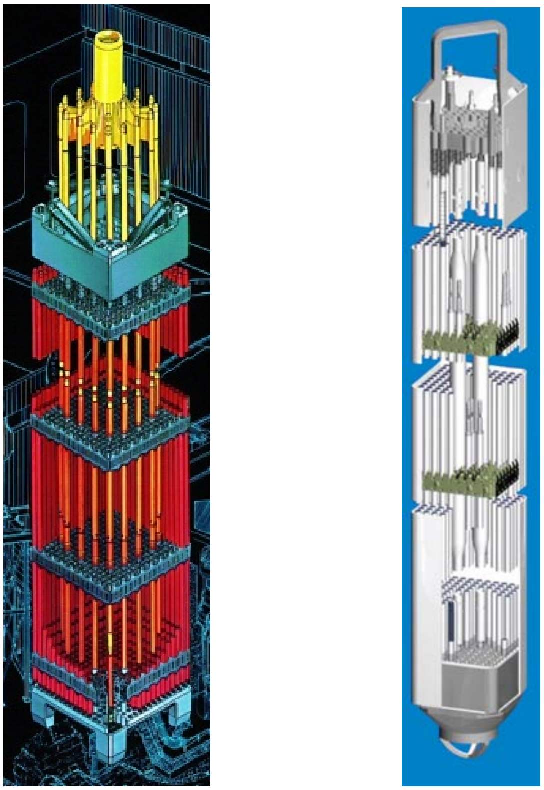

Figure 1. Typical Pressurized-Water Reactor (left) and Boiling-Water Reactor Fuel Assemblies
Courtesy Westinghouse (left)
http://gepower.com/prod_serv/products/nuclear_energy/en/downloads/gnf2_adv_poster.pdf (right)

Following discharge from the reactor, used fuel is initially stored in racks under water in pools up to 40 feet deep (see Figure 2). During this period of wet storage, some degradation may be detectable, although it is typically minimal.

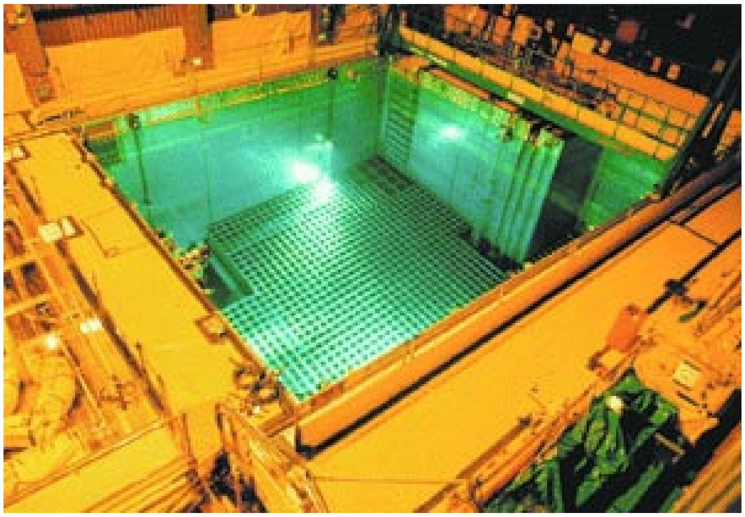

Figure 2. Typical Used-Fuel Storage Pool
(from http://www.nrc.gov/waste/spent-fuel-storage/pools.html)

Before the pools at a nuclear power plant are filled to their licensed capacity, the operator needs to provide additional storage capacity so that the power plant can continue operating. Thus, many utilities have built dry-storage facilities (referred to as Independent Spent Fuel Storage Installations, or ISFSIs) on their sites. These installations are large parking-lot-type concrete pads with protective fencing and continuous security surveillance. The fuel may be stored vertically in metal or concrete casks or horizontally in modular concrete storage facilities. The fuel inside concrete dry-storage casks is in bolted or welded canisters that are loaded in the spent-fuel pool and transferred to the ISFSI in an on-site transfer cask. Similar canisters are used for fuel that is stored in horizontal storage modules and may be used to contain fuel in metal storage casks, although some metal casks contain the fuel in open baskets without an inner canister.

Figure 3 shows concrete casks storing used fuel at the Connecticut Yankee Nuclear Power Plant site. Figure 4 shows typical components that constitute a concrete storage-cask system, including the multipurpose canister (MPC) and the vents that provide the airflow to cool the canister. Metal casks that contain the fuel in open baskets do not have the same ventilation arrangement but typically have external heat-transfer fins to assist with the cooling.

Figure 3. Independent Spent-Fuel Storage Installation
Photo from NAC International, Inc. with permission

Before used fuel is loaded from a pool into a canister, the canister is lowered into the storage pool inside a shielded transfer cask. If the fuel is to be loaded into a metal cask with an open basket, the cask is lowered into the pool with the basked installed. Following loading of the fuel, a lid is installed and the cask is removed from the pool. The water is drained from the cask and, if a canister is being used, the water also is drained from the canister. The lid then is bolted or welded in place, and operations to dry the fuel are started.

This process typically involves several cycles of alternately applying a vacuum and backfilling the canister, or the cask, with helium. During the periods when the vacuum is applied, the fuel rods lose much of their cooling and the temperature of the fuel rises. The temperature rise enhances drying, but the temperature has to be controlled below predetermined limits to prevent thermal stresses that could result in cladding damage. Once the drying process is completed, the canister or cask is pressurized with helium, both to provide improved heat transfer and to minimize the potential for fuel degradation during subsequent storage.

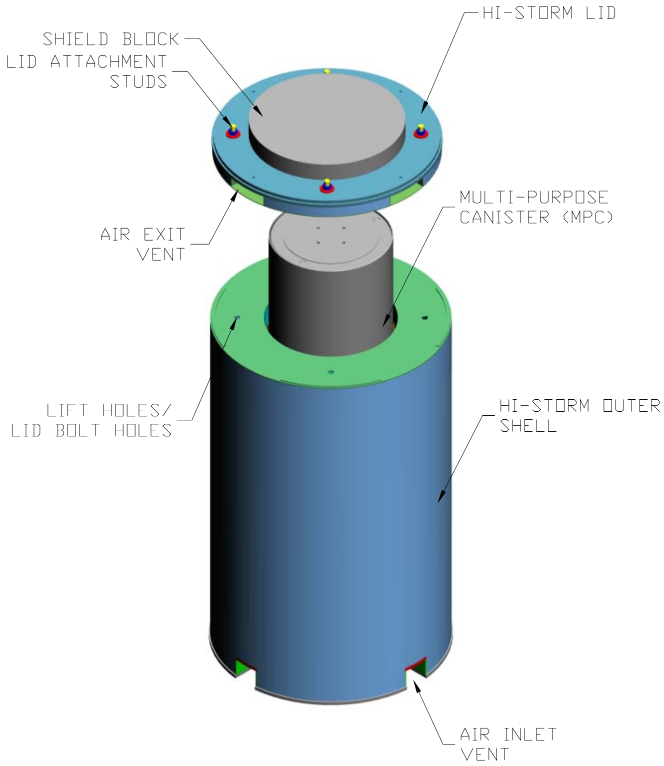

Figure 4. Typical Vertical Dry Cask-Storage System
Graphic Courtesy of Holtec International, Inc.

Findings

This review finds that fuel rods discharged from nuclear power plants are typically in good condition with only a very small percentage of rods having cladding defects. Early references reported that less than 0.04 percent of fuel rods failed, while later plant records indicate that the failure rate has decreased to less than 0.0005 percent for more recently discharged fuel. During preparation for transfer to dry-storage facilities, failed fuel assemblies are loaded into specifically designed compartments in the canisters or metal casks, separate from intact fuel assemblies.

The fuel-drying process is not perfect. After drying, residual water remains in unknown amounts that can affect subsequent internal degradation processes. The vacuum-drying heat cycles can change the nature of the hydrogen in the cladding and stress the fuel.

According to the literature review, the fuel, the dry-storage system components (canister, cask, etc.), and the concrete foundation pad may all degrade during dry storage. Some degradation mechanisms may be active during the early years of dry storage, while different mechanisms may be active at the lower temperatures that would be expected during extended storage.

The most significant potential degradation mechanisms affecting the fuel cladding during extended storage are expected to be those related to hydriding, creep, and stress corrosion cracking. These mechanisms and their interactions are not yet well understood. New research suggests that the effects of hydrogen absorption and migration, hydride precipitation and reorientation, and delayed hydride cracking may degrade the fuel cladding over long periods at low temperatures, affecting its ductility, strength, and fracture toughness. High-burnup fuels tend to swell and close the pellet-cladding gap, which increases the cladding stresses and can lead to creep and stress corrosion cracking of cladding in extended storage. Fuel temperatures will decrease in extended storage, and cladding can become brittle at low temperatures.

Only limited references were found on the inspection and characterization of fuel in dry storage, and they all were performed on low-burnup fuel after 15 years or less of dry storage. Insufficient information is available yet on high-burnup fuels to allow reliable predictions of degradation processes during extended dry storage, and no information was found on inspections conducted on high-burnup fuels to confirm the predictions that have been made. The introduction of new cladding materials for use with high-burnup fuels has been studied primarily with respect to their reactor performance, and little information is available on the degradation of these materials that will occur during extended dry storage. Consequently, without any data for predicting how aging affects the fuel condition over longer storage periods, vendors model the condition of high-burnup used fuel currently in storage on the basis of the limited series of examinations of fuel that have been performed to date. These also form the basis for predicting the behavior of used fuel during extended dry storage and normal handling and transport of used fuel and in the event of transportation accidents.

As noted above, one of the main deterrents to corrosion of the fuel cladding and the canister or metal cask internals during extended dry storage is the presence of helium. If the helium leaks and air is allowed to enter the canister or cask, this, together with the moisture in the air, can result in corrosion of the fuel cladding, the canister, and the cask. However, although provision is made to monitor the pressure of the helium during extended storage in bolted canisters, there is currently no means of confirming the presence of helium in welded containers or casks, nor is there a requirement for periodically inspecting the integrity of the closure welds for defects. If these storage systems were inspected for weld defects and/or tested for helium periodically, this would allow welded containers and casks with leaks to be repaired and refilled with helium.

During extended dry storage, degradation mechanisms also act on the outside of canisters, on storage casks (concrete or steel), and on modular concrete facilities as well as on the storage pads. The effect of these degradation mechanisms will depend on the environmental conditions at the specific location, on diurnal and seasonal temperature variations, and on the presence of corrosive agents and moisture in the air. The review identified references to general metal and concrete deterioration mechanisms and modeling, but none included the information necessary to predict the degradation of dry-storage canisters, casks, or concrete structures during extended storage.

Given the temperature dependence of many of the degradation mechanisms described above, accurately predicting how the used fuel and canister temperatures will change over extended dry storage is important. During this review, however, little information was found on detailed thermal modeling during the period of extended dry storage.

Regardless of the length of storage, used fuel eventually will have to be moved from the reactor sites, either to off-site interim storage facilities or to used-fuel processing facilities for recycling or for management of waste. Transportation regulations are largely focused on the integrity of the transportation casks, which contain the used fuel, and maintaining subcriticality of the fuel. The primary goal is to ensure that the cask does not fail in the event of a transportation accident, with the potential for release of radioactive materials. The regulations require that under both normal and accident conditions, the transportation cask and its contents are capable of meeting stringent performance specifications that include maintaining geometric configuration of the fuel to certain limits, largely for criticality control, and to address concerns about external radiation levels.

If the fuel degrades during extended storage, it could be susceptible to damage from the vibration and shocks encountered during transport operations. The consequences may include release of fission-product gases into the canister or the cask interior, which must be contained during a transportation accident.

Used-fuel transportation casks are designed to withstand a series of transportation accidents without release of radioactive materials. Figure 5 shows a full-scale crash test performed by Sandia National Laboratory in 1977 in which a locomotive traveling at approximately 80 miles per hour was crashed broadside into a used-fuel transportation cask. In this test, the cask and the dummy fuel inside it performed in accordance with the regulatory requirements.

Figure 5. Spent-Fuel Crash Test
http://www.sandia.gov/recordsmgmt/ctb1.html

Upon reaching the interim storage location, the repository site, or other processing facility, the used fuel may have to be handled, and the integrity of the fuel following the transportation and handling operations may not be known with confidence. If the fuel is to be processed instead of being placed in a repository after transportation, the casks and used-fuel canisters will need to be opened and the fuel removed. Before this is done, consideration will need to be given to the condition of the fuel, and a means will need to be available for determining whether the fuel has failed. This may require opening the cask in a hot cell as opposed to the more traditional spent-fuel pool. Following extended storage, the integrity of the used fuel after transportation cannot be ensured because some long-term degradation processes are uncertain and transportation-accident loading predictions for aged fuels have not been fully validated.

Review of the relevant technical sources used as the basis for this report has shown the following:

- Little data are publicly available on the behavior of high-burnup fuel during dry storage or on its behavior during subsequent handling and transportation. No information is available on the behavior during dry storage of the more advanced materials now being used for fuel cladding and fabrication of fuel-assembly structural components.
- The physical state of the cladding when fuel is placed into dry storage is not currently well characterized. There may be zones of physical weakness and, in some cases, the cladding may be close to failure. Normal handling of fuel assemblies, off-normal occurrences, and accident events would then be more likely to result in additional damage to fuel rods.
- Cladding-degradation mechanisms, their interactions with each other, and the expected behavior of cladding after extended dry storage are not well understood. Also not well understood are some of the conditions that affect these degradation mechanisms, such as the changes in fuel temperatures that will occur over time and the amount of residual water present after drying.
- At the low temperatures expected to be reached during extended dry storage, and even in the presence of air, used-fuel-pellet material oxidizes at a very slow rate. Even if a gross breach occurs and fuel-pellet material is released from the fuel cladding, it will not oxidize to powder easily or quickly. Consequently, if fuel material is released inside the canister or cask, containing and repackaging it safely once the canister or cask is opened should not present any undue problem. Fission-product gases also would be released inside the canister or the cask. If the canister or cask were to be opened, they would need to be dealt with by the ventilation system in the fuel-handling facility.
- Corrosion mechanisms will cause degradation of the metal components of dry-storage systems during extended dry-storage periods: for example, the outer surfaces of fuel canisters. Consequently, establishing an effective regular inspection and maintenance program is important.
- In order to develop a better understanding of how the conditions of fuel and storage systems change during extended dry-storage, it will be important to establish a program for inspecting and characterizing the physical condition of used fuel and dry-storage systems over time. This will reduce the uncertainty in predicting the future state and behavior of the used fuel and the storage-system components during subsequent operations.
- Several concrete deterioration and rebar-corrosion mechanisms are known to cause degradation of reinforced concrete in dry-storage systems, including the storage pad. Consequently, establishing a regular inspection and maintenance program for these systems is important.
- Some plausible off-normal and accident scenarios for the handling and transport of used-fuel casks have not been fully evaluated. Performing full-scale testing of transport packages to demonstrate the behavior of both the package and the fuel may be beneficial. At a minimum, validation of computer models using scaled tests should be carried out.

However, the performance of some components, such as bolts and welds, are particularly difficult to scale. Consequently, if scaled tests are performed, additional testing of full-scale components may be needed to verify that the performance of these components is being modeled correctly.

- There are security risks associated with the dry storage of used fuel, and the risks will likely change with time. These risks and how they change need to be addressed using a risk-informed process that considers the probability of the risks and the potential consequences. This process then should be used as the basis for determining what action, if any, is needed to provide the necessary level of security during extended dry-storage periods.

Research and Development Recommendations

On the basis of this review, we recommend that a number of research and development programs be implemented. They are focused primarily on improving the understanding of key fuel-degradation mechanisms and increasing confidence in the projection of the behavior of the used fuel and storage systems during extended dry-storage periods and subsequent transportation of the fuel. The intention is to prevent problems that may otherwise be encountered during later fuel-handling operations following transportation of used fuel to disposal or processing facilities. The recommended research programs investigate the following issues:

- Understanding the mechanical cladding behavior and fuel-cladding degradation mechanisms potentially active during extended dry storage, including those that will act on the materials introduced in the last few years for fabrication of high-burnup fuels
- Understanding and modeling the time-dependent conditions that affect aging and degradation processes, such as temperature profiles, in situ material stresses, quantity of residual water, and quantity of helium gas
- Modeling of age-related degradation of metal canisters, casks, and internal components during extended dry storage
- Inspection and monitoring of fuel and dry-storage systems to verify the actual conditions and degradation behavior over time, including techniques for ensuring the presence of helium cover gas
- Verification of the predicted mechanical performance of fuel after extended dry storage during cask and container handling, normal transportation operations, fuel removal from casks and containers, off-normal occurrences, and accident events
- Design and demonstration of dry-transfer fuel systems for removing fuel from casks and canisters following extended dry storage

Conclusions

The technical information currently available, together with the experience gained to date in the dry storage of spent fuel, demonstrates that used fuel can be safely stored in the short term and then transported for additional storage, processing, or repository disposal without concern. However, additional information is required to demonstrate with similarly high confidence that used fuel can be stored in dry-storage facilities for extended periods without the fuel degrading to the extent that it may not perform satisfactorily during continued storage and subsequent transportation.

Consequently, the Board recommends that a number of research and development programs be implemented to demonstrate that used fuel can be stored safely in dry-storage facilities for extended periods. Research alone will not be sufficient. Because the experience base for extended dry storage of used fuel is short and the credible degradation phenomena are several and not robustly predictable in a quantitative sense, an in-service inspection and maintenance program appears to be the only way of implementing long-term dry storage of used nuclear fuel. The technical details of such an in-service inspection program obviously will depend on the desired safety objectives of extended dry storage. Consequently, a practical engineering approach that is based on the observational method and periodic assessments will likely be required to provide an adequate safety basis in addition to what can be learned from targeted scientific investigations.

The regulations concerning dry storage of used fuel do not currently address storage for extended periods. There also is some inconsistency between the regulations that apply to dry storage and those that apply to transportation, and how to meet both sets of regulations is unclear. It would be helpful in managing extended dry storage of used fuel if the regulations were revised as an integrated set and based on a risk assessment for safety significance and consequence. In addition, the Board thinks that the regulatory requirements related to physical security and terrorist threats also should be reviewed on a risk-informed basis using potential consequence analysis and integrated with the storage and transportation regulations.

At this point, the nuclear waste management policy of the United States is unclear, and the result is that used fuel will be stored at reactor sites for longer than originally foreseen. It is thus essential that the appropriate research and development programs and monitoring and inspection programs are implemented as a matter of priority to demonstrate that used fuel can be stored safely for extended periods and then transported and handled as part of a future waste management program.

Abstract

The U.S. Department of Energy (DOE) has announced that it plans to terminate the Yucca Mountain repository project. This decision, if upheld, means that used nuclear fuel stored at nuclear plant sites, and used fuel and high level waste produced by the Department of Energy at its sites, will remain at those sites for decades longer than expected. This report summarizes the current state of knowledge regarding long-term (60 to 120 years) and very long-term (greater than 120 years) dry storage and subsequent transportation of used commercial nuclear fuel. Based on this review, the findings of the technical literature suggest that active degradation mechanisms continue during the storage period on all aspects of the storage system that are not fully understood. Consequently, the condition of the used fuel upon transport cannot be reliably predicted. This is especially true for high-burnup fuels currently in use and the new cladding and fuel assembly structural materials now being introduced. Additionally, once the used fuel is eventually shipped to either a repository or a waste processing facility, it is not clear that the used fuel will arrive undamaged, so the possibility exists that special precautions will need to be taken prior to opening the sealed canisters most likely in hot cells.

This report recommends several areas of research that are needed to improve the confidence in long-term storage and subsequent transportation and handling. The recommendations also include the development of technologies that can verify the presence of helium in the storage system since its function over the long-term is to assist in cooling the fuel and inhibit certain corrosion mechanisms. Periodic inspections of canisters may be required, as may full-scale tests of transportation casks, should the mathematical modeling and semi-scale cask tests not provide adequate data to ensure cask integrity during severe accidents. Additionally, the current Nuclear Regulatory Commission regulations need to be reviewed for long-term storage and integrated using a risk informed consequence assessment of both storage and transportation incorporating security concerns.

The 1987 Amendments to the Nuclear Waste Policy Act direct the U.S. Nuclear Waste Technical Review Board to independently review the technical validity of U. S. Department of Energy (DOE) activities regarding the management and disposal of used nuclear fuel and high-level radioactive waste. The information in this report was prepared to inform Congress about important technical issues associated with extended storage of used nuclear fuel and its subsequent transportation. The information also will be used by the Board to conduct its review of DOE activities, when appropriate. The Board expects the report will also be valuable in informing the Blue Ribbon Commission on America’s Nuclear Future and other interested parties on these issues.

Introduction

The U.S. Department of Energy (DOE) has announced that it plans to terminate the Yucca Mountain repository project and has convened the Blue Ribbon Commission on America's Nuclear Future to evaluate alternative approaches for meeting the federal responsibility to manage and ultimately dispose of used nuclear fuel and high-level radioactive waste from both commercial and defense activities.[1] An implication of this course of action is that more commercial spent or used nuclear fuel (CSNF) will be stored for longer periods than previously anticipated. It is unknown when or whether currently stored CSNF will be transported, and if so, if the subsequent destination will be an interim storage site, a reprocessing plant, or a federal waste repository. Under current U.S. law, at some point (DOE) will come into possession of all CSNF.

Purpose

The U.S. Nuclear Waste Technical Review Board (NWTRB) is tasked by the Amendments to the Nuclear Waste Policy Act of 1982 to review the technical validity of DOE's activities regarding CSNF and high-level waste. To prepare to evaluate DOE's technical involvement in handling, storing, and transporting CSNF, the NWTRB initiated this background study to review readily available reports and research articles that pertain to these subjects. This report identifies relevant technical concerns related to extending safe dry storage and then transportation of CSNF over the long-term, which for the purposes of this report is defined to be 60 to 120 years, and very long-term, a storage period of 120 years and longer. It also identifies information needs and potential areas of research to better understand and ensure continued integrity, during handling, storage, and transport of CSNF and subsequent disposition either for disposal or processing

Scope

This report reviews available public literature on storage and handling of used nuclear fuel related to the safety[2] of long (greater than 60 years) and very-long term (greater than 120 years) of dry storage and subsequent transportation of U.S. commercial-owned used fuel after long storage periods. The present scope does not encompass long-term wet storage of fuel, DOE-owned used fuel and high-level radioactive waste., . It does not analyze physical security[3] threats but makes some observations in this area. This study is a generalized qualitative analysis of the subject rather than one that delves into quantitative details and analyses such as the specifics of particular cask designs or site characteristics.

U.S. Background Information

The nuclear fuel for typical U.S. commercial reactors consists of pellets of uranium dioxide (UO_2) surrounded by a zirconium alloy cladding that is formed into closed tubes (fuel rods). These fuel rods are between 11.5 and 14.5 feet (3.5 and 4.5 meters) in length and are bundled together into fuel assemblies, each weighing between 600 and 1500 pounds (275 to 685 kilograms).

Nuclear fuel discharged from a nuclear plant reactor is termed “spent” or “used”[4] fuel and is intensely radioactive. One year after a fuel assembly has been discharged, the dose rate measured at a one-meter distance is on the order of 1,000 sieverts per hour – which would give a lethal dose to a person in less than a minute.[5] This radioactivity also creates heat (mostly from the radioactive decay of ^{137}Cs and ^{90}Sr). CSNF is initially stored in racks in a pool of water (wet storage), which provides radiation shielding and cooling. When a used-fuel pool nears storage capacity, the older fuel may be moved to dry-storage systems on concrete storage pads. Under the current U.S. regulations, CSNF may be transferred to dry storage only after at least five-years residence time in a pool.

Dry-storage systems are designed to safely contain and protect CSNF, dissipate the heat generated by radioactive decay of the CSNF, shield workers and the public from radiation emitted by the CSNF, and prevent significant degradation of the fuel. Storing the fuel for several decades substantially decreases the radioactive decay heat. U.S. dry-storage systems for commercial CSNF are licensed by the U.S. Nuclear Regulatory Commission (NRC) according to Title 10, Part 72 of the Code of Federal Regulations (10 CFR 72)[6] and are known as “Independent Spent Fuel Storage Installations” (ISFSIs).[7] The NRC is the Federal agency responsible for regulating the safety and security of CSNF storage. Approximately 22 percent of domestic CSNF is in dry storage at 44 plant sites.[8] Figure 6 shows both the existing and likely future locations of CSNF dry storage.[9]

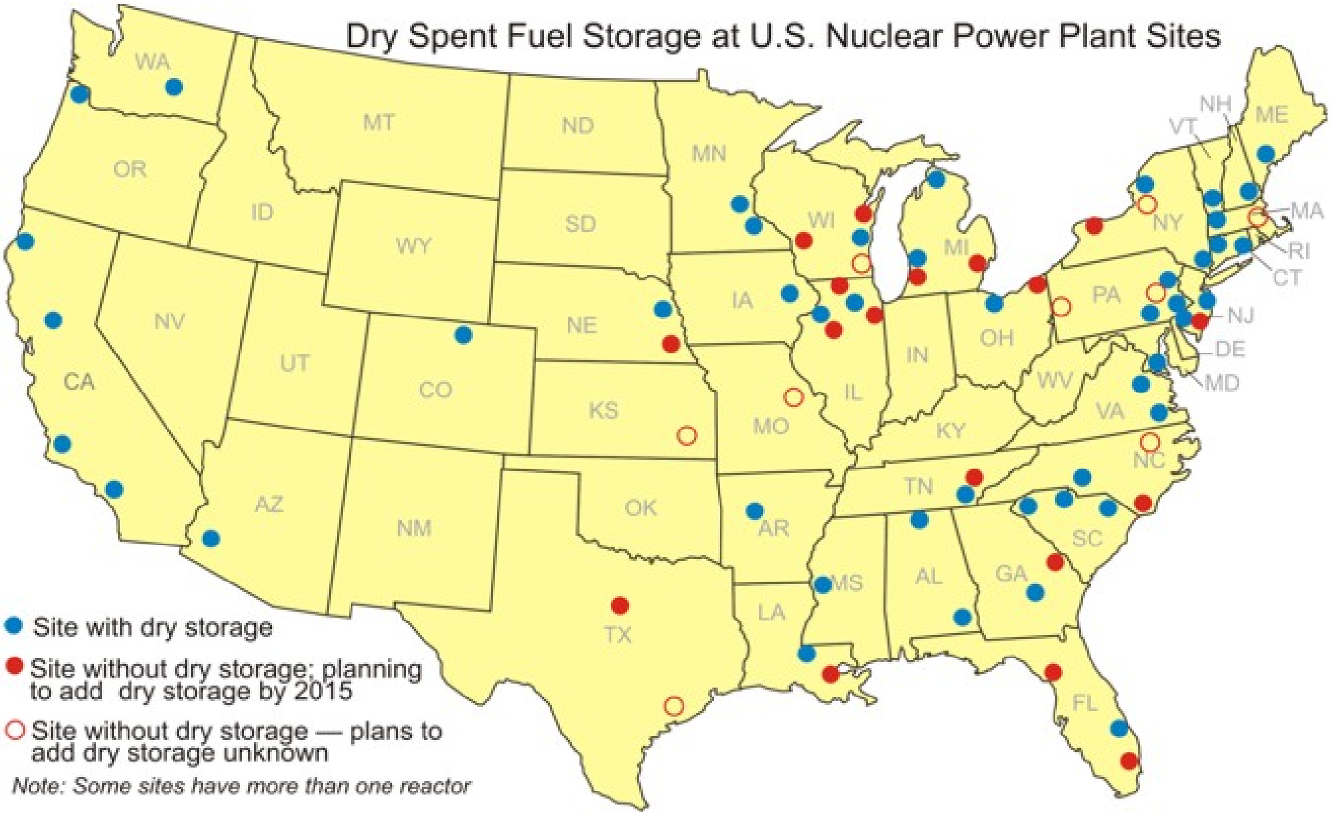

Figure 6. U.S. Commercial Dry Used-Fuel Storage Facilities

Under current regulations, NRC licenses CSNF dry-storage systems initially for 20-year periods. However, NRC recently authorized an exemption to the regulation and renewed the license for a dry-storage system at the Surry plant in Virginia for a 40-year term for a total of 60 years.[10] On September 15, 2009, NRC proposed changing the periods for initial licensing and license renewals for dry-storage systems from 20 years to 40 years.[11]

The reality of dry-storage requirements today is much different from what early nuclear scientists and engineers anticipated. In the original plans, used nuclear fuel was to be removed from reactors and reprocessed within several years. However, repository siting and licensing delays subsequently led to the need for dry storage. When the ISFSI systems were originally designed, it was assumed by the designers, owners, and regulators that CSNF would be sent offsite within several decades for disposal at a repository when it became available. Accordingly, the technical bases for dry-storage designs did not extend to long periods, say more than 60 years.

Since storage – regardless of it duration – is by definition temporary, capability to handle the CSNF properly at the end of the storage period must be provided.[12] All CSNF in dry storage will eventually need to be transported to reprocessing, a different storage facility, or permanent disposal. As the length of time needed to retain used fuel in storage increases, the CSNF dry-storage inventory will continue to increase. Furthermore, when it is time to finally handle and transport the stored CSNF, it is important to understand the condition of the used fuel after long- and very-long term storage.

Overview

- Section 1 provides a brief description of licensed dry-storage systems in use in the U.S.
- Sections 2 and 3 summarize the regulatory requirements for dry storage and handling of CSNF, and, transportation of CSNF, respectively.
- Subsection 4.1 describes the typical physical state of used fuel-pellets and cladding before dry storage, describes the components of dry-storage systems, and notes current trends.
- Subsection 4.2 describes the three operational phases of dry storage and then describes how key environmental conditions are expected to change with time.
- Subsection 4.3 lists the important events, conditions, and triggering events including normal and off-normal conditions, accident conditions (man-made and natural phenomena), and physical security or hazards.
- Subsection 4.4 provides technical background information on all significant degradation and aging processes affecting the various dry-storage elements and components.
- Subsection 4.5 addresses the risk of criticality during dry storage.
- Sections 5 and 6 evaluate and summarize the key findings regarding long- and very-long term storage and handling of CSNF, then transportation of CSNF, respectively.

- Section 7 summarizes what has been concluded regarding incomplete data, and the needs for additional data collection, new analysis, research, and monitoring.
- Section 8 presents the overall conclusions and recommendations for this study.

1. Characteristics of U.S. Dry-Storage Systems

Two basic types of storage systems are in use in the United States: bare-fuel (thick-walled or metal shielded) casks and canister-based systems consisting of a (thin-walled) canister inside a (thick-walled) cask or storage module (concrete-shielded). In bare-fuel cask storage systems, used-fuel assemblies are placed directly into a basket that is integrated into the cask itself and then the cask is sealed using two bolted lids. Most bare-fuel systems are for storage only.

In canister-based storage systems, used-fuel assemblies are loaded into baskets integrated into a thin-wall (typically 0.5-inch or 13-mm thick) stainless steel cylinder, referred to as a canister. The canister is sealed with two welded lids. Most canister-based systems are designed to be dual-purpose canisters (DPC); that is, the canister can be stored or transported if it is placed within a suitable storage or transportation overpack. The dry storage systems loaded today are all of the DPC type. Originally, some of the early designs were multi-purpose canisters (MPC) that were designed for storage, transportation and direct disposal, but DOE was unable to define suitable disposal criteria. Thus, these canisters are limited to storage and transport only while direct disposal of the canisters is still a possibility. Dual purpose designs bypass the need for fuel to be returned to the reactor pool for repackaging into a transport package. For storage the DPCs or MPCs are placed in either a cylindrical overpack system made up of concrete and steel, or a concrete vault-type overpack system. The overpack protects the canister against external man-made events and external natural phenomena, and functions as a shielding/thermal barrier. The overpack is typically closed with a bolted lid. Shown in Figure 7 is a typical canister-based storage system.

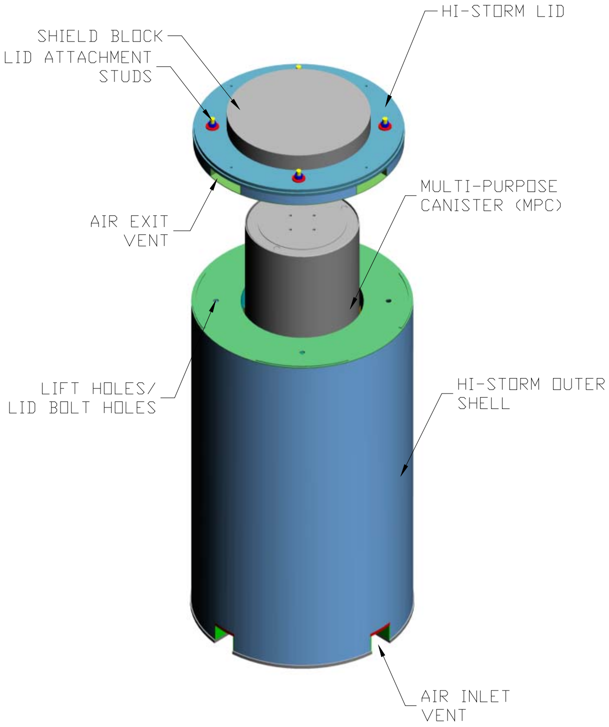

Figure 7. Typical Canister-Based Storage System
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Dry storage first began in the U.S. in the mid 1980s with storage-only systems. The first dry cask storage facility was established at the Surry Nuclear Power Plant (Virginia) in 1986. Cost considerations and interest in dual-purpose systems are driving the storage industry away from all-metal cask designs and toward concrete canister designs.[13] The dry cask storage system is placed as a free-standing structure on a reinforced concrete pad supported by an engineered soil foundation. The principal characteristics of U.S. dry cask storage systems are summarized in Tables 1 and 2, those of U.S. multi-purpose systems are summarized in Tables 3 and 4, and those of U.S. transportation-only casks are summarized in Table 5. The footnotes for all three tables are provided in Table 2.

The NRC approves standardized used-fuel storage cask design systems in accordance with the 10 CFR 72 regulations.[6] The owners of the nuclear power plants can then use the NRC-approved dry cask storage system at the same site as the nuclear power plant under a 10 CFR 72 general license. Private companies and owners of the nuclear power plants can also obtain a 10 CFR 72 site-specific license to store nuclear waste using one of the NRC-approved dry cask storage systems. NRC has approved 21 models of dry-storage systems.[14]

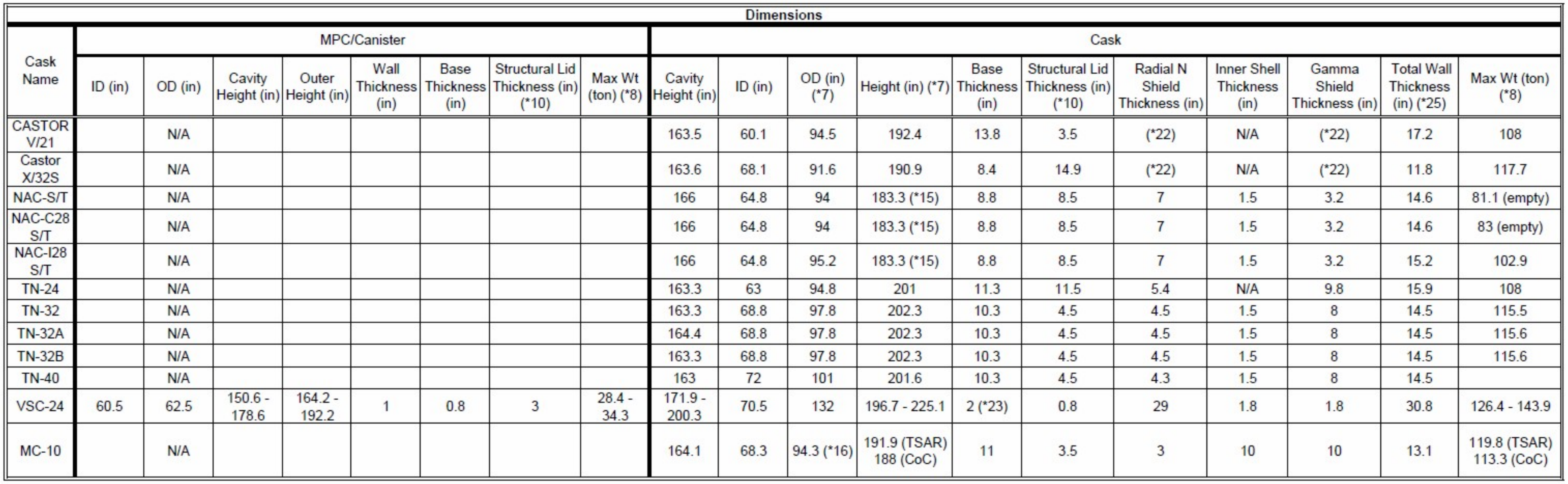
According to the Nuclear Energy Institute, currently more than 13,500 metric tons heavy metal (MTHM) of used fuel are in dry storage at 44 sites, and by 2020 more than 30,000 MTHM used fuel will be in dry storage at 73 sites.[15] This represents more than 1,200 loaded storage casks today, and reaching more than 2,400 casks by 2020. With no new plant construction there is projected to be about 9,300 casks and canisters in dry storage by 2055 (about 100,000 MTHM).[16] Operating nuclear power plants typically generate about 20 metric tons of used fuel each year. Each dry-storage cask of modern design can typically store between about 10 and 18 metric tons of used fuel; the oldest cask designs had a capacity limit of about half this amount.

A variety of dry-storage systems are in use today. Tables 1 and 2 provide detailed design information and dimensions for storage-only canister and cask systems. These systems were not designed to be transported under current regulations and so it is expected that this fuel will need to be placed in transportation casks for shipment. This requires either dry transfer or placing the used fuel back into a pool. Tables 3 and 4 provide detailed design information for multi or dual purpose casks. These canister based systems can be transported directly. Table 5 provides detailed design information on transportation casks that can carry bare and canistered used fuel.

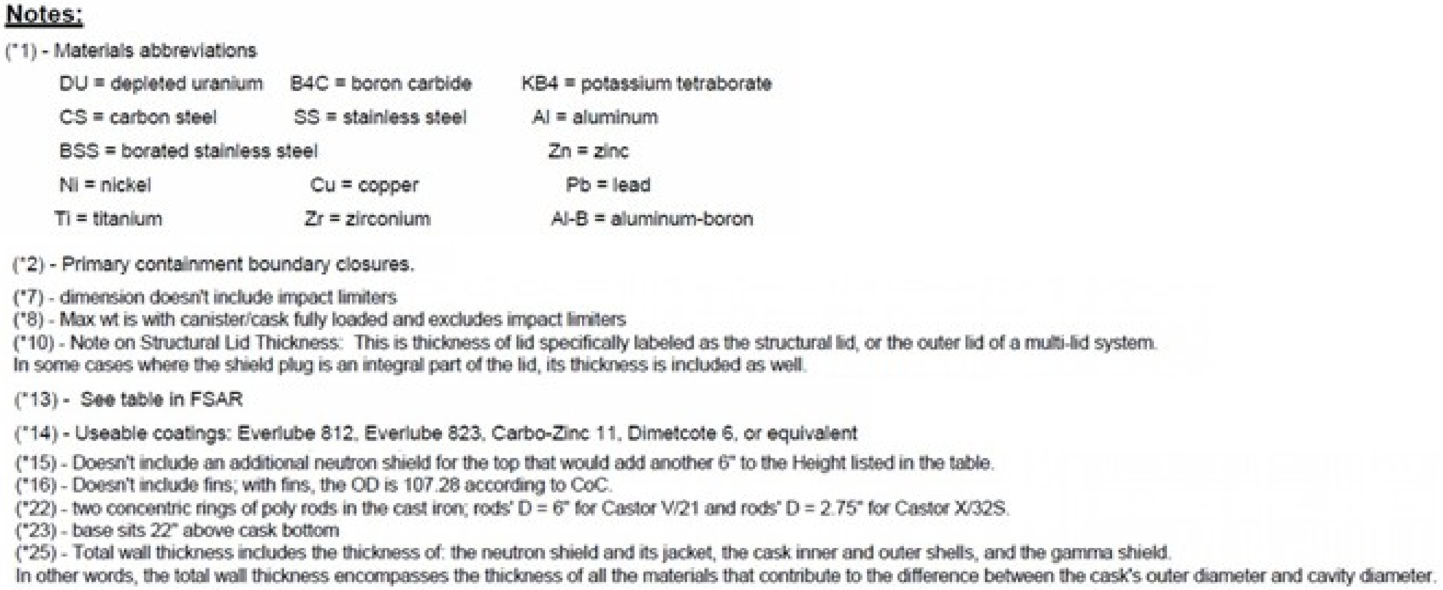
Table 1. Characteristics of Dry Cask *Storage-Only Systems* in the United States – Basic Design Information[17]

		Cask Design				Fuel Characteristics					
Designation	**Status/Users**	**Description**	**Materials (*1)**	**Cask Outer Diameter (*7)**	**Closures (*2)**	**Type**	**Max. Init. Enrichm't, %**	**Max. Burnup (GWD/MTU)**	**Min. Cool Time (yr)**	**Cask Heat Output (kW)**	**Unique Contents**
CASTOR V/21	CERT-1000 Surry (SNM-2501)	vertical metal cask on concrete pad	Cast Iron, SS, BSS, Ni Coating, Polyethylene, Epoxy Resin Coating, Elastomer Seals	94.5"	bolted	21 PWR	3.5 3.7	35 40	5 6	21.1	
Castor X/32S	Under Review (72-1028)	vertical carbon steel cask with borated aluminum (NAM) rods on concrete pad	CS, SS, Al, Polyethylene, Al-B4C Composite, Flame-sprayed Al, Inconel, Viton, BROCOPHAN, Silicon Sealant	91.6"	1) Bolted inner and outer 2) Bolted inner, welded outer	32 PWR	5.0	45	2.9 - 9.5	32	BPRAs
NAC-S/T	CERT-1002	vertical metal cask on concrete pad	SS, Pb, Al, Cu, NS4FR, Boral	94"	bolted	26 PWR	3.3	35	5	26	
NAC-C28 S/T	CERT-1003	vertical metal cask on concrete pad; contains consolidated FA's - fuel rods from 56 PWR FAs consolidated into 28 canisters	SS, Pb, Al, Cu, NS4FR, Boral	94"	bolted	28 PWR	3.5	35	10	20	
NAC-I28 S/T	Surry approved TSAR (SNM-2501) No Certificate	vertical metal cask on concrete pad	SS, Pb, Al, Cu, NS4FR, Boron Impregnated Al, Silicone Foam	95.2"	bolted	28 PWR	1.9	35	10	15.6	BPRAs, TPDs
TN-24	CERT-1005	vertical metal cask on concrete pad	CS, Al, Zn/Al & Ti/Al-Oxide Coatings, Cu, Borated Polyester Resin, Borated SS, SS, Al, Viton O-rings, Polypropylene	94.8"	bolted	24 PWR	3.5	35	5	24	
TN-32	CERT-1021	vertical metal cask on concrete pad	CS, Al Coating, Al, SS, Borated Polyester Resin, Borated Aluminum, Viton, Polypropylene	97.8"	bolted	32 PWR	4.05	45	7 (*13)	32.7	BPRAs, TPAs
TN-32A	CERT-1021	vertical metal cask on concrete pad	CS, Al Coating, Al, SS, Borated Polyester Resin, Borated Aluminum, Viton, Polypropylene	97.8"	bolted	32 PWR	4.05	45	7 (*13)	32.7	BPRAs, TPAs
TN-32B	CERT-1021	vertical metal cask on concrete pad	CS, Al Coating, Al, SS, Borated Polyester Resin, Borated Aluminum, Viton, Polypropylene	97.8"	bolted	32 PWR	4.05	45	7 (*13)	32.7	BPRAs, TPAs
TN-40	Prairie Is. (SNM-2506)	vertical metal cask on concrete pad	CS, Al, Zn/Al Coating, Borated Polyester Resin, Viton, Polypropylene, SS, Boral	101"	bolted	40 PWR	3.85	45	10	27	BPRAs, TPDs
VSC-24	CERT-1007	vertical concrete overpack w/ metal sealed basket & transfer cask	Ceramic Tiles, CS, SS, Concrete, Viton O-rings, Rx-277, Coatings (*14)	132"	welded canister, bolted overpack	24 PWR	4.2	51.8 45	5 (*13)	24	BPRAs, TPAs
MC-10	Ineel/Surry approved TSAR (SNM-2501) CERT-1001(NOTE: Letter in for termination of CoC)	vertical metal cask on concrete pad	SS, CS, Low Alloy Steel, NS3, Al, B4C, Ni, Boral, Inconel, Carbo-Zinc8 Coating, Ethylene, Propylene, Polyacrylate	94.3"	welded (TSAR says bolted & 3rd cover can be welded)	24 PWR	3.7	35	10	13.5	BPRAs, TPDs

Table 2. Characteristics of Dry Cask *Storage-Only Systems* in the United States – Detailed Dimensions, Notes for All Tables



Dimensions																			
	MPC/Canister								Cask										
Cask Name	ID (in)	OD (in)	Cavity Height (in)	Outer Height (in)	Wall Thickness (in)	Base Thickness (in)	Structural Lid Thickness (in) (*10)	Max Wt (ton) (*8)	Cavity Height (in)	ID (in)	OD (in) (*7)	Height (in) (*7)	Base Thickness (in)	Structural Lid Thickness (in) (*10)	Radial N Shield Thickness (in)	Inner Shell Thickness (in)	Gamma Shield Thickness (in)	Total Wall Thickness (in) (*25)	Max Wt (ton) (*8)
CASTOR V/21		N/A							163.5	60.1	94.5	192.4	13.8	3.5	(*22)	N/A	(*22)	17.2	108
Castor X/32S		N/A							163.6	68.1	91.6	190.9	8.4	14.9	(*22)	N/A	(*22)	11.8	117.7
NAC-S/T		N/A							166	64.8	94	183.3 (*15)	8.8	8.5	7	1.5	3.2	14.6	81.1 (empty)
NAC-C28 S/T		N/A							166	64.8	94	183.3 (*15)	8.8	8.5	7	1.5	3.2	14.6	83 (empty)
NAC-I28 S/T		N/A							166	64.8	95.2	183.3 (*15)	8.8	8.5	7	1.5	3.2	15.2	102.9
TN-24		N/A							163.3	63	94.8	201	11.3	11.5	5.4	N/A	9.8	15.9	108
TN-32		N/A							163.3	68.8	97.8	202.3	10.3	4.5	4.5	1.5	8	14.5	115.5
TN-32A		N/A							164.4	68.8	97.8	202.3	10.3	4.5	4.5	1.5	8	14.5	115.6
TN-32B		N/A							163.3	68.8	97.8	202.3	10.3	4.5	4.5	1.5	8	14.5	115.6
TN-40		N/A							163	72	101	201.6	10.3	4.5	4.3	1.5	8	14.5	
VSC-24	60.5	62.5	150.6 - 178.6	164.2 - 192.2	1	0.8	3	28.4 - 34.3	171.9 - 200.3	70.5	132	196.7 - 225.1	2 (*23)	0.8	29	1.8	1.8	30.8	126.4 - 143.9
MC-10		N/A							164.1	68.3	94.3 (*16)	191.9 (TSAR) 188 (CoC)	11	3.5	3	10	10	13.1	119.8 (TSAR) 113.3 (CoC)



Notes:

(*1) - Materials abbreviations

DU = depleted uranium   B4C = boron carbide   KB4 = potassium tetraborate
CS = carbon steel   SS = stainless steel   Al = aluminum
BSS = borated stainless steel   Zn = zinc
Ni = nickel   Cu = copper   Pb = lead
Ti = titanium   Zr = zirconium   Al-B = aluminum-boron

(*2) - Primary containment boundary closures.

(*7) - dimension doesn't include impact limiters
(*8) - Max wt is with canister/cask fully loaded and excludes impact limiters
(*10) - Note on Structural Lid Thickness: This is thickness of lid specifically labeled as the structural lid, or the outer lid of a multi-lid system. In some cases where the shield plug is an integral part of the lid, its thickness is included as well.

(*13) - See table in FSAR

(*14) - Useable coatings: Everlube 812, Everlube 823, Carbo-Zinc 11, Dimetcote 6, or equivalent

(*15) - Doesn't include an additional neutron shield for the top that would add another 6" to the Height listed in the table.
(*16) - Doesn't include fins; with fins, the OD is 107.28 according to CoC.
(*22) - two concentric rings of poly rods in the cast iron; rods' D = 6" for Castor V/21 and rods' D = 2.75" for Castor X/32S.
(*23) - base sits 22" above cask bottom
(*25) - Total wall thickness includes the thickness of: the neutron shield and its jacket, the cask inner and outer shells, and the gamma shield. In other words, the total wall thickness encompasses the thickness of all the materials that contribute to the difference between the cask's outer diameter and cavity diameter.

Table 3. Characteristics of *Multi-Purpose Dry-Storage Systems* in the United States – *Basic Design Information*

		Cask System Design					Fuel Characteristics					
Designation	**Status**	**Description**	**Materials** (*1)	**Cask Outer Diameter** (*7)	**Impact Limiter Design**	**Closures** (*2)	**Type**	**Max. Init. Enrichm't,%**	**Max. Burnup (GWD/MTU)**	**Min. Cool Time (yr)**	**Max. Cask Heat Output (kW)**	**Unique Contents**
Holtec HI-STAR 100 (transport)	Certificate (71-9261)	vertical metal overpack w/ dual-purpose canister & 3 basket assemblies	SS, CS, Al, Holtite-A Paints, Boral	96"	Aluminum honeycomb in stainless steel skin	welded canister, bolted overpack	24 PWR 68 BWR 68F BWR	3.8 - 4.6 PWR 4.0,5.0,5.5 BWR 4.0,5.5 BWR (68F)	24.5-44.1 PWR 24.5-39.1 BWR 30 BWR (68F) (*3)	7-24 PWR 8-18 BWR 18 BWR (68F) (*3)	9.0 - 16.9 PWR 12.2 - 16.2 BWR 7.82 BWR (68F)	Zr/SS clad fuel, failed fuel BWR MOX fuel, Thoria rods, Thoria FAs w/ Sb-Be neutron source
Holtec HI-STAR 100 (storage)	Certificate (72-1008)	vertical metal overpack w/ dual-purpose canister & 3 basket assemblies	SS, CS, Al, Holtite-A Paints, Boral	96"		welded canister, bolted overpack	24 PWR 68 BWR 68F BWR	3.8 - 4.6 PWR 4.0,5.0,5.5 BWR 4.0,5.5 BWR (68F)	28.3 - 42.1 PWR 22.5 - 37.6 BWR 30 BWR (68F) (*3)	5-15 PWR 5-18 BWR 18 BWR (68F) (*3)	13.8 - 19.0 PWR 6.46 - 18.5 BWR 7.82 BWR (68F)	Zr/SS clad fuel, failed fuel BWR MOX fuel, Thoria rods, Thoria FAs w/ Sb-Be neutron source
Hi-STORM 100 (storage)	Certificate (72-1014)	3 vertical concrete casks w/ variant for high seismic applications, dual purpose canister w/ 7 baskets, 2 transfer casks	SS, CS, Concrete, Boral, Holtite-A, Pb, Al, Water (in transfer cask only)	132.5"		welded canister, bolted overpack	24 PWR 32 PWR 68 BWR 68F BWR	3.8 - 5 PWR 4, 5, 5.5 PWR 4, 5, 5.5 BWR (*18)	40 - 58 PWR 30 - 47.2 PWR 22.5 - 53.9 BWR 22.5 - 30 (68F) (*18)	5 - 15 PWR 5 - 18 BWR (*18)	17 - 28.1 PWR 28.7 PWR 12.4 BWR 7.82 (68F) (*18)	PWR control components, BWR MOX, Thoria rods, PWR & BWR damaged fuel and fuel debris
TN-68 (transport)	Certificate (71-9293)	metal cask for transport of BWR FAs	CS, Al, SS, Al-B Alloy, Pb, Boralyn, Borated Polyester Resin (NOT NS4FR)	98"	Balsa wood and redwood in carbon steel shell w/ an aluminum spacer	bolted	68 BWR	3.7	40	10 - 28 (*18)	21.2	
TN-68 (storage)	Certificate (72-1027)	vertical metal cask on concrete pad	CS, Al, SS, Polypropylene, Al-B Alloy, Pb, Boralyn, Borated Polyester Resin (NOT NS4FR)	98"		bolted	68 BWR	3.7	40	10 - 12	21.2	
TN-FSV (*19)	Certificate (71-9253)	Steel and lead shielded shipping cask for Fort St. Vrain (FSV) and Peach Bottom-1 (PB1)	SS, Pb, Silicone/Butyl O-rings, Al, DU	31"	Balsa wood and redwood in stainless steel shell	bolted	6 FSV 1 Peach Bottom Unit 1 (PB1)	93.5 (1.4kg U w/ 11.3kg Th) FSV 93.15(0.25kg U w/ 1.5kg Th) PB1	70 - FSV 73 - PB1	1600 days FSV 27 PB1	0.36 FSV 0.12 PB1	HTGR FAs from FSV or PB1, or fuel parts from PB1
NAC-MPC (storage)	Certificate (72-1025)	vertical metal/concrete overpack w/ dual-purpose canister, transfer cask & 3 basket assemblies	SS, CS, Al, NS4FR/NS-3, Pb, Concrete, Boral, Silicone Foam	128"		welded canister, bolted overpack	34,36 PWR (YC) 24,26 PWR (CY)	3.5 - 4.97 YC 2.95 - 4.61 CY	32, 36 YC 30 - 43 CY	8.0 - 24 YC 6 CY	8.74 - 12.5 YC 17.5 CY	intact & failed YC and CY fuel, flow mixer, thimble plug assemblies, Reactor control cluster assemblies
NAC-STC (transport)	Certificate (71-9235)	metal overpack w/ dual-purpose canister, transfer cask & 3 basket assemblies(*4)	SS, Al, NS4FR, Pb, Cu, Boral/Talbor, Viton Seals	99"	Balsa wood and redwood in stainless steel shell	bolted	26 PWR 36 PWR (YC) 24,26 PWR (CY)	4.2, 4.5 PWR 3.5 4.94 YC 3.93 - 4.61 CY	45 PWR 32, 36 YC 38, 43 CY	5 - 15 PWR 8 - 19 YC 10 CY	22.1 PWR 12.5 YC 7.7 - 17 CY	failed, Yankee Class,Connecticut Yankee fuel, GTCC waste
NAC-UMS (transport)	Certificate (71-9270)	metal overpack w/ dual-purpose canister & transfer cask(*5)	SS, Al, NS4FR, Pb, Cu, Boral	92.9"	Balsa wood and redwood in stainless steel shell	welded canister, bolted overpack	24 PWR 56 BWR	4.2 PWR 4.0 BWR	45 PWR 45 BWR	5 PWR 6 BWR	20 PWR 16 BWR	intact and damaged PWR FAs and Maine Yankee fuel debris and GTCC waste, BPRs
NAC-UMS (storage)	Certificate (72-1015)	vertical metal/concrete overpack w/ dual-purpose canister, transfer cask (*5)	SS, CS, Al, NS4FR, Pb, Concrete, Boral, Ni Coating	136"		welded canister, bolted overpack	24 PWR 56 BWR	4.2 PWR 4.0 BWR	30 - 45	5 - 15 PWR 5 - 26 BWR	23	thimble plugs, BPRs, site specific fuels, Maine Yankee fuel, damaged PWR and site specific rods/FAs

Table 4. Characteristics of *Multi-Purpose Dry-Storage Systems* in the United States – Detailed Dimensions

Cask Name	Dimensions																		
	MPC/Canister								Cask										
	ID (in)	OD (in)	Cavity Height (in)	Outer H (in)	Wall Thickness (in)	Base Thickness (in)	Structural Lid Thickness (in) (*10)	Max Wt (ton) (*8)	Cavity Height (in)	ID (in)	OD (in) (*7)	Height (in) (*7)	Base Thickness (in)	Structural Lid Thickness (in) (*10)	Radial N Shield Thickness (in)	Inner Shell Thickness (in)	Gamma Shield Thickness (in)	Total Wall Thickness (in) (*25)	Max Wt (ton) (*8)
HI-STAR 100 (transport)	67.4 (67.5)	68.4 (68.5)	178.5 PWR 178 BWR	190.3	0.5	2.3	9.5 PWR 10 BWR	44.5	191.1	68.8	96	203.1	6	6	4.4	2.5	6	13.6	121.5
HI-STAR 100 (storage)	67.4 (67.5)	68.4 (68.5)	178.5 PWR 178 BWR	190.3	0.5	2.3	9.5 PWR 10 BWR	44.5	191.1	68.8	96	203.1	6	6	4.4	2.5	6	13.6	121.5
Hi-STORM 100 (storage)	67.4	68.4	178.5 178	190.5	0.5	2.5	9.5 10	45	191.5 191.5 203	73.5	131 132.5 132.5	239.5 231.3 242.8	24.5 19 19	24 20.8 20.8	26.8 27.5 27.5 (*20)	1.3	26.8 27.5 27.5 (*20)	28.8 29.5 29.5	180
TN-68 (transport)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	178	69.5	98	197.3	9.8	5	6	1.5	6	14.3	113.7
TN-68 (storage)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	178	69.5	98	215	9.8	5	6	1.5	6	14.3	114.5
TN-FSV	16.6	17.6	187.3	195.3	0.5	2	1.5	2.1	199	18	31	207	5.5	2.5		1.1	3.4 (*21)	6.5	20.5
NAC-MPC	69.4	70.6	113.5 YC 142 CY	122.5 YC 151.8 CY	0.6	1.0 YC 1.8 CY	3	27.4 YC 32.9 CY	140.8 YC 169.3 CY	79	128	160 YC 190.6 CY	2.0 YC 4.0 CY (*9)	1.5	21 (*20)	3.5	21 (*20)	24.5	77.5 YC 93 CY
NAC-STC	69.4	70.6	113.5 YC 142 CY	122.5 YC 151.8 CY	0.6	1.0 YC 1.8 CY	3	27.4 YC 32.9 CY	165	71	99	193	13.7	5.3	5.5	1.5	3.7	14	121.6
NAC-UMS (transport)	65.8	67.1	163.3 - 180.0	175.1 - 191.8	0.6	1.8	3	35.3 - 38	192.5	67.6	92.9	209.3	10.3	6.5	4.5	2	2.8	12.7	115.3 - 118.6
NAC-UMS (storage)	65.8	67.1	163.3 - 180.0	175.1 - 191.8	0.6	1.8	3	35.3 - 38	189.3 - 206	74.5	136	209.2 - 225.9	2.0 (*9)	1.5	28.2 (*20)	2.5	28.2 (*20)	30.7	146.2 - 156.1
Std.NUHOMS-24P/PHB (storage)	65.9	67.2	167 173	186.2	0.6	8.7 5.9	1.3	39											
Std.NUHOMS-24PT2 (storage)	65.9	67.2	167 173	186.5 192.5	0.6	8.7 5.9	1.2	42.2 41											
Std. NUHOMS-52B (storage)	65.9	67.2	177.5	195.9	0.6	8.5	1.3	40											
Std. NUHOMS-61BT (storage)	66.3	67.3	179.3	196	0.5	7.8	1.3	44.3											
NUHOMS MP-197 (transport)									197	68	91.5	208	6.5	4.5	4.6	1.3	3.3	11.8	118.6
NUHOMS MP-187 (transport)									187	68	92.5	201.5	8	6.5	4.3	1.3	4	12.3	125
TS125 Fuel Solutions (transport)	64.8 W21 64.8 W74	66.0 W21 66.0 W74	163 - 180 W21 173 W74	182.3, 192.3 W21 192.3 W74	0.6 W21 0.6 W74	4.4 - 8.5 W21 8.5 W74	2	36.3 - 40.4 W21 37.7, 38.8 W74	193	67	94.2	210.4	11.3	6	6	1.5	3.3	13.6	124.8
FuelSolutions (storage) (*8)	64.8 W21 64.8 W74	66.0 W21 66.0 W74	163 - 180 W21 173 W74	182.3, 192.3 W21 192.3 W74	0.6 W21 0.6 W74	4.4 - 8.5 W21 8.5 W74	2	36.3 - 40.4 W21 37.7, 38.8 W74	193 183	73	138	230 220	18	14.3	30.5 (*20)	2	30.5 (*20)	32.5	167 160.5

Table 5. Characteristics of *Transportation-Only Casks* in the United States – Basic Design Information and Detailed Dimensions

		Cask Design					Fuel Characteristics					
Designation	**Status/ Certificate Holder**	**Description**	**Materials (*1)**	**Cask Outer Diameter (*7)**	**Impact Limiter Design**	**Closures (*2)**	**Type**	**Max. Init. Enrichm't, %**	**Max. Burnup (GWD/MTU)**	**Min. Cool Time (yr)**	**Cask Heat Output (kW)**	**Unique Contents**
IF-300	Certificate (71-9001) Chem-Nuclear Systems, Inc.	stainless steel encased, DU shielded cask with 3 basket designs	SS, DU, Cu, Metallic Ring, Water-Ethylene-Glycol Mix	64"	32 Stainless steel impact fins, 16 extend 8" from cask end and 16 extend 6" from cask end	bolted	6,7 PWR 17,18 BWR	4.0 PWR 4.0 PWR 4.0 BWR 4.25 BWR 4.25 PWR	45 PWR 35 PWR 35 BWR 45 BWR 45 PWR	5 PWR 120 days PWR 120 days BWR 4 BWR 5 PWR	11.72 (listed in CoC as 40000 BTU/hr)	see CoC
NLI-1/2	Certificate (71-9010) NAC International, Inc.	stainless steel encased, DU-water-lead shielded shipping cask with 4 configurations (one has inner canister)	SS, DU, Pb, (Borated) Water-Ethylene-Glycol Mix, Ag-plated O-ring, Elastomer O-ring, Al, Boral	36.8"	Balsa wood structure in aluminum weldment	bolted	1 PWR FA 2 BWR FAs 25 PWR rods 18 BWR rods	3.7 PWR FAs 2.65 BWR FAs 4.9 PWR rods 5.0 BWR rods	34 PWR FAs 34 BWR FAs 60 PWR rods 75 BWR rods	150 days PWR FA 120 days BWR FA 150 days PWR, BWR rods	10.6 PWR, BWR FAs 1.65 PWR rods 4.0 BWR rods	neutron source components, see CoC for others
NAC-LWT	Certificate (71-9225) NAC International, Inc.	steel-encased, lead shielded cask	SS, Pb, Metallic and Teflon O-rings, Ethylene Glycol/Water Solution of 1w/o Boron	39.2"	Aluminum honeycomb	bolted	1 PWR FA 2 BWR FAs 25 rods	3.5,3.7 PWR FA 4.0 BWR FA 5 rods	35 PWR FAs 30 BWR FAs 60, 80 PWR rods 80 BWR rods	2 PWR, BWR FAs 150 days PWR rods 150 - 270 days BWR rods	2.5 PWR FA 2.2 BWR FAs 1.41, 2.3 PWR rods 2.1 BWR rods	Damaged & intact PWR and BWR rods, see CoC for others
GA-4	Certificate (71-9226) General Atomics	cask with square cavity for 4 PWR spent fuel assemblies	SS, DU, B4C, Aqueous Propylene Glycol - 1%B (KB4) Solution, Ethylene Propylene O-ring	39.8"	Aluminum honeycomb in stainless steel structure	bolted	4 PWR	3.15	35 45	10 15	2.47	control rods, non-fuel assembly hardware

Dimensions																		
	MPC/Canister								Cask									
Cask Name	ID (in)	OD (in)	Cavity Height (in)	Outer Height (in)	Wall Thickness (in)	Base Thickness (in)	Structural Lid thickness (in) (*10)	Max Wt (ton) (*8)	Cavity Height (in)	ID (in)	OD (in) (*7)	Height (in) (*7)	Base Thickness (in)	Structural Lid Thickness (in) (*10)	Radial N Shield Thickness (in)	Inner Shell Thickness (in)	Gamma Shield Thickness (in)	Total Wall Thickness (in) (*25)	Max Wt (ton) (*8)
IF-300 (71-9001)			N/A						180.3 BWR 169.5 PWR	37.5	64	192.4 BWR 182.2 PWR	6.5	5.5 BWR 6 PWR	7.1	0.5	4	13.3	70
NLI-1/2 (71-9010) (*12) (*24)	12.6	13.1	177.9	184.3	0.3	1.4	5	1.8	189.9 Config A 178.2 others	13.4	36.8	197.4	6	1.5	5	0.5	4.9	11.7	23.2 - 23.9
NAC-LWT (71-9225) (*12)			N/A						178	13.4	39.2	199.8	10.5	11.3	5	0.8	5.8	12.9	24.2
GA-4 (71-9226) (*11)			N/A						167.3	18.2 (flats) 24.6 (corners)	39.8	187.8	9.5	11	6.0 (flats) 2.8 (corners)	0.4	2.7	10.8 (flats) 7.6 (corners)	25.5

Dry-storage systems are loaded by placing them directly into the pool, which is a 40 foot deep pool of water used to cool and provide radiation shielding from the used-fuel elements. Any damaged used fuel-rods are first placed into special containers before being placed in the basket.[18] After a system is loaded, it is removed from the pool and placed on a flat surface capable of supporting its weight, its outer surfaces are decontaminated, and the lid is bolted or welded in place. Liquid water and moisture are removed from the system by a vacuum drying system or using the forced helium dehydration system.

Following moisture removal, the system is backfilled with an inert gas (typically helium), the small openings in the lid for drying and inert gas filling are sealed, the integrity of the seals is checked, and then a final cover lid is bolted or welded in place. The loaded canisters are then moved to the dry-storage facility on the property of the reactor site, typically on open concrete pads within a protected area of the plant. Loading of a single system can take up to one week and is dry storage-system dependent. The drying process is one of the longest steps in the loading process.[19]

Dry-storage systems are passive systems requiring minimal human attention. Radioactive decay processes in the used fuel generate heat. Passive cooling of the used fuel occurs through a combination of heat conduction through solid materials, natural air convection through storage system air ducts, and thermal radiation. Nuclear criticality (a self-sustaining nuclear chain reaction) control is provided by the basket (a lattice structure) that holds the used-fuel assemblies within individual compartments in the cask, by excluding moderator (water) from the storage system, and by the low reactivity of the used fuel. The basket may contain boron-doped metals to absorb neutrons. Dry-storage systems are designed for specific types of used fuel and for specific fuel burnups.

All bare-fuel dry-storage systems in use in the United States store the used fuel vertically (Figure 8 shows a typical schematic). Most canister-based systems also are designed for vertical storage such as the Holtec International HI-STORM 100 dry-storage system (Figure 9 and Figure 10). Used fuel also is stored horizontally in a concrete module or vault such as the NUHOMS HSM system (see Figure 12). Modern dry-storage casks and canisters hold about 10-18 metric tons of used fuel, which is equivalent to about 24-32 pressurized water reactor (PWR) assemblies or 56-68 boiling water reactor (BWR) assemblies. Although the dimensions vary among manufacturers, fuel types, and amounts of fuel stored, the casks are typically about 19 feet (6 meters) in height, 8 feet (2.5 meters) in diameter, and weigh 100 to 180 tons or more when loaded.

Heat generated by the CSNF in a loaded system is transported from the assemblies to the basket structure by convection of the inert gas, conduction where the assemblies touch the basket structure, and radiation. Heat then moves from the basket structure to the inner cask or canister wall by the same mechanisms. Metal cask bodies typically have external heat transfer fins and are cooled by air flow past the fins. The fins are required for casks containing fairly recent CSNF to meet the temperature limit requirements of the used fuel. Some metal cask designs have an external metal shell (open at top and bottom) as weather protection. Canisters within concrete casks are cooled by natural convection of air introduced from air inlets at the bottom of the concrete. In heat from the hot canister wall warms air entering via inlets and gives it buoyant force, inducing a natural air circulation upward and out of the upper exhaust vents in a self-sustained manner.

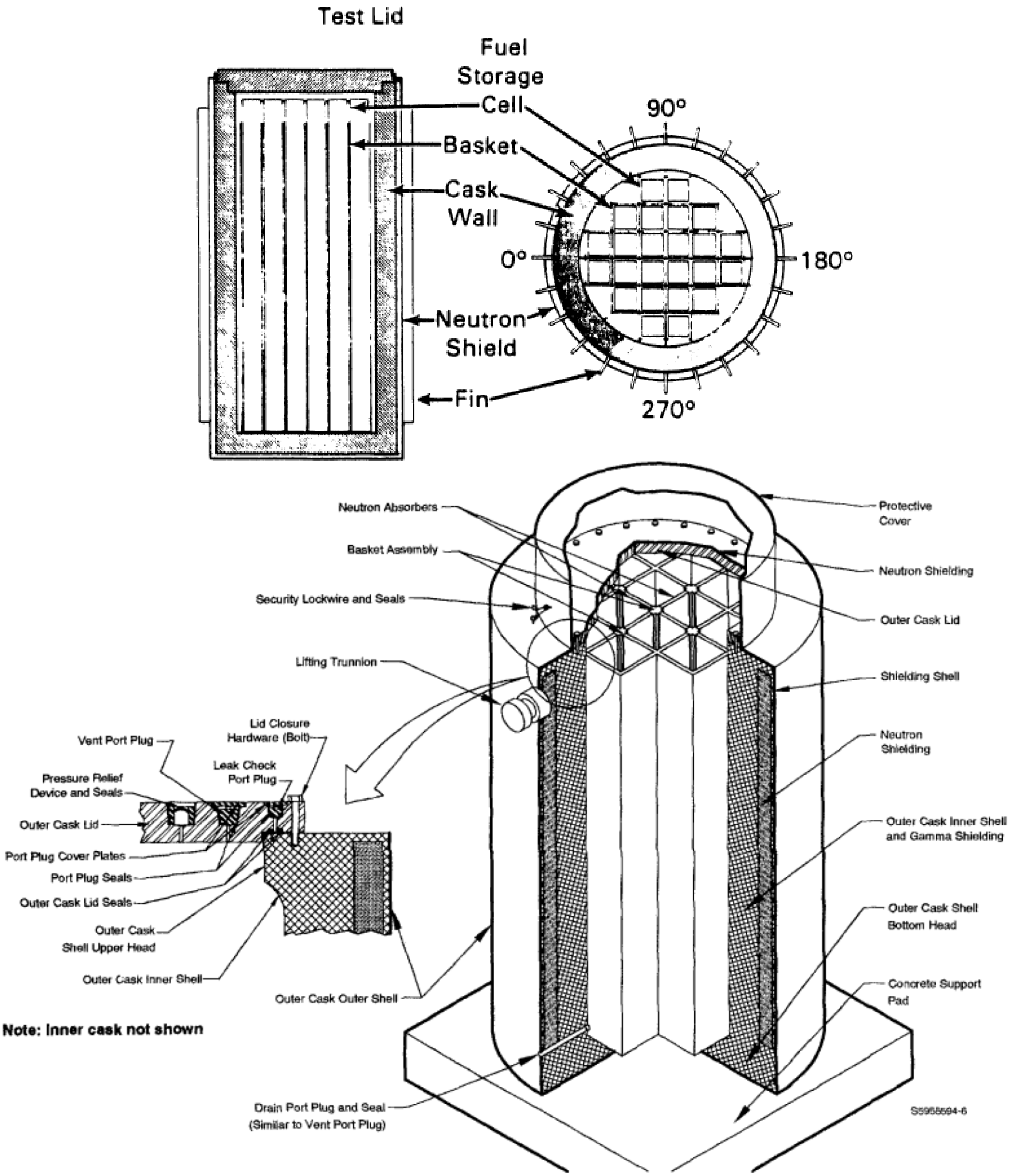


Figure 8. Vertical Metal-Shielded Dry Cask System
Used with permission of Holtec International

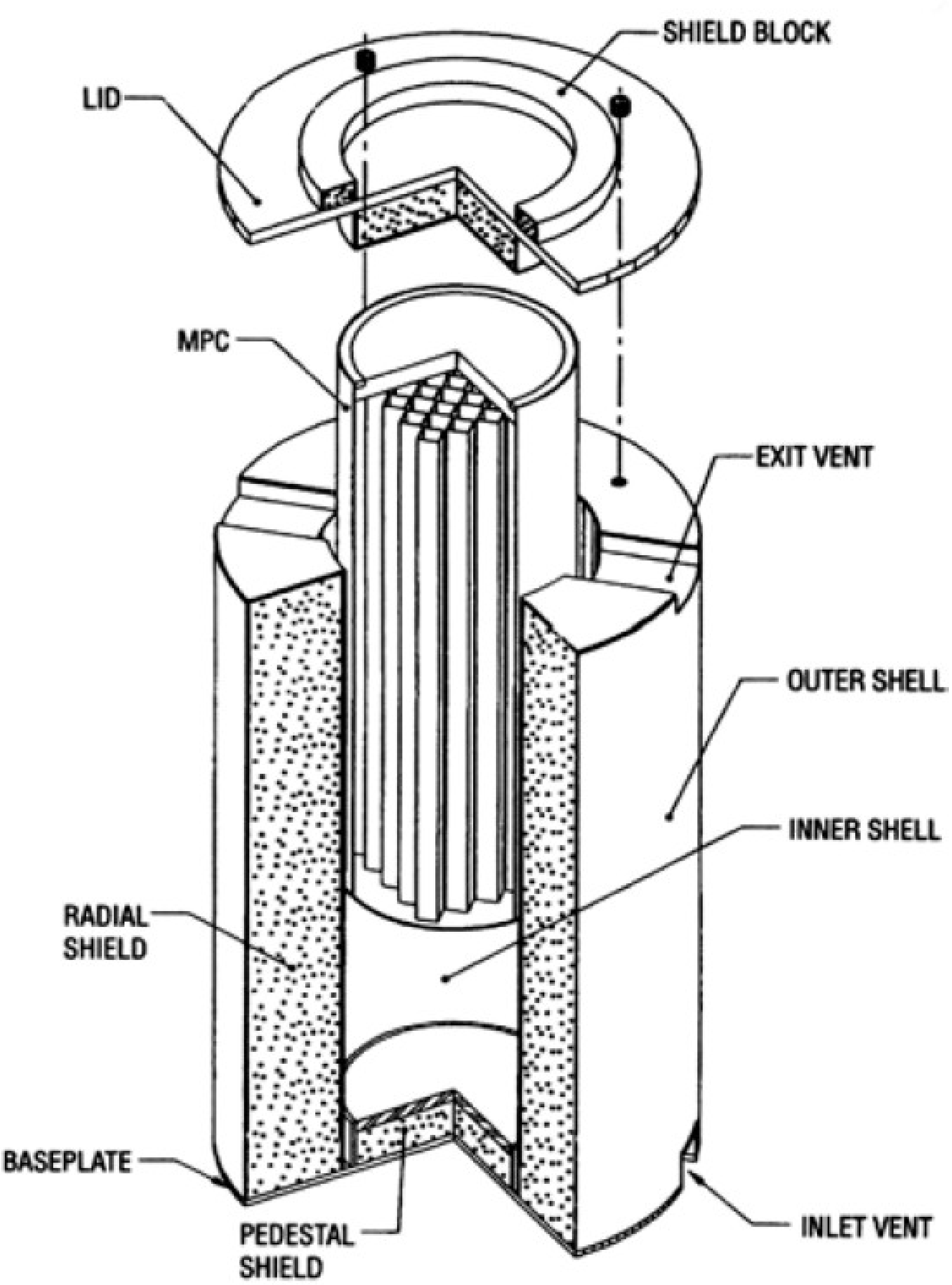


Figure 9. Vertical Welded-Canister Concrete-Shielded Dry-Storage System

Figure 10. Holtec Vertical Storage Casks
Used with permission of Holtec International, Inc.

The HI-STORM 100 dry cask storage system consists of a sealed metallic canister, referred to as the Multi-Purpose Canister (MPC), contained within a steel-concrete-steel overpack. Different MPCs are available to accommodate a wide variety of used-fuel assemblies and each has different internal baskets to accommodate distinct fuel characteristics. The overpack is constructed of two cylindrical steel shells filled with unreinforced concrete. NAC has also developed vertical storage systems based on the MPC concept. Shown on Figure 11 is a typical NAC storage system.

Figure 11. Independent Spent Fuel Storage Installation at Connecticut Yankee
Used with permission of NAC International, Inc.

A third type of storage system was developed by Nutech, which is now being sold by Transnuclear (Figure 12 andFigure 13). This is a horizontally loaded and stored canister system contained in air-cooled concrete vaults.

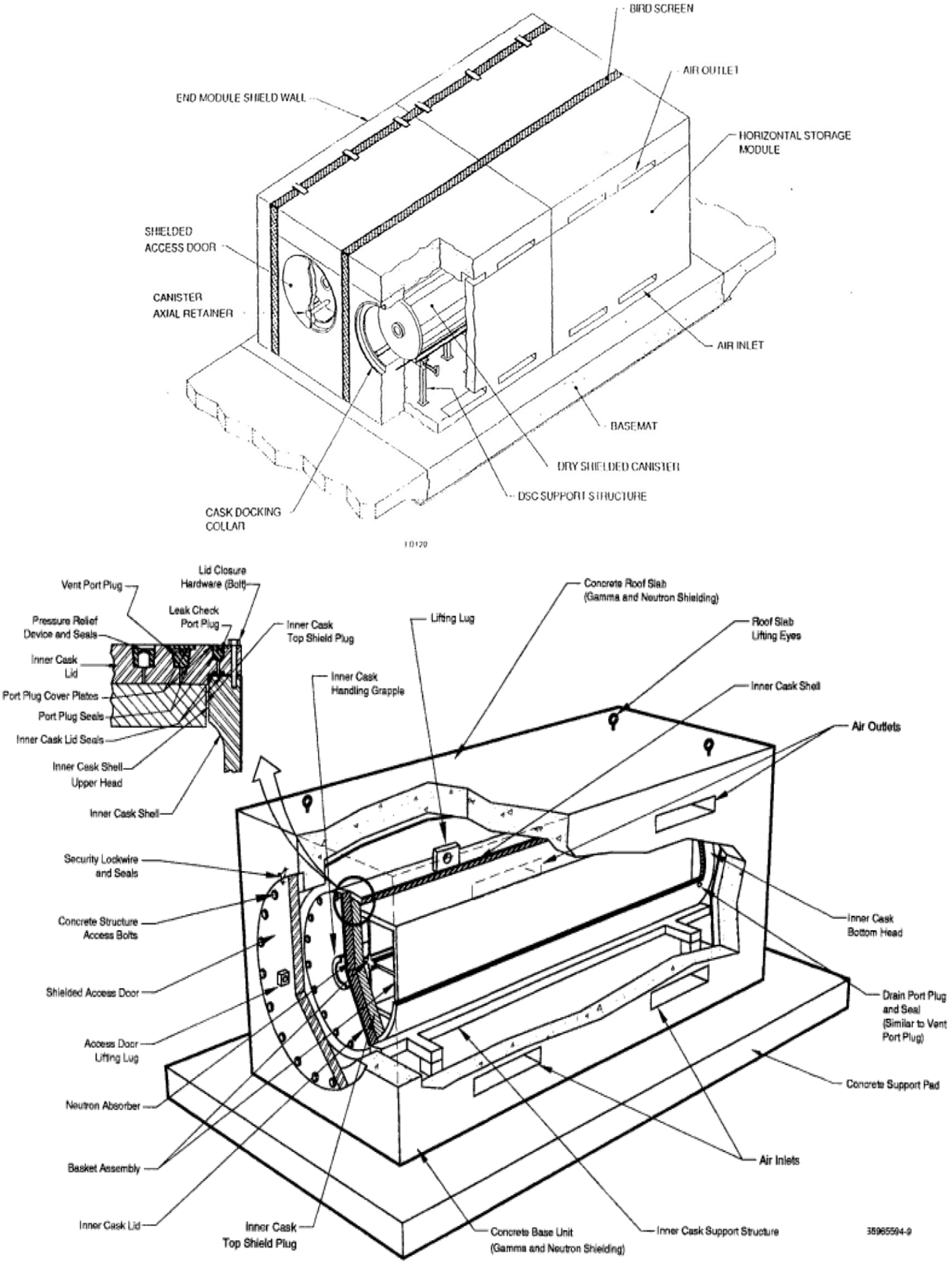


Figure 12. Horizontal Concrete-Shielded Dry-Storage System
Used with permission of Transnuclear

Figure 13. Transnuclear Horizontal ISFSI at San Onofre
Used with permission of Transnuclear, Inc.

Bare-fuel casks use seals between the cask and the lids, and helium pressure between the primary (inner) and secondary (outer) lids is monitored. Failure of the secondary lid seal has occurred on several casks leading to fuel retrieval, inspection, and resealing.[20] Canisters in canister-based systems are sealed by partial-penetration welds between the canister wall and the primary and secondary canister lids.[21] One of the welds is helium-leak tested after welding; helium pressure between the lids is not monitored after the canister-based system has been moved to its storage location, however. Loss of helium by diffusion through a metal lid or canister shell is expected to be insignificant in 100 years – or even 1,000 years.[22] More details about the dry-storage system components can be found in subsection 4.1.3.

2. Requirements for Dry Storage and Handling

The NRC approach for licensing dry-storage systems to ensure the safe storage and handling of used fuel is based on demonstrating that the used fuel must be easily retrievable, not experience gross ruptures, and satisfy certain performance objectives. The evaluation of these objectives can be complicated and they are described more fully below. For the purpose of this report, the evaluation of these performance objectives (the U.S. regulation requirements) will be assumed to be reducible to the concurrent satisfaction of three criteria: fuel-rod integrity, fuel assembly geometric arrangement, and canister integrity.

(1) *Fuel-rod integrity* means no significant numbers of individual fuel rods suffer a "gross breach," as defined below.

(2) *Fuel assembly geometric arrangement* is maintaining the original geometric arrangement of fuel assemblies and lifting points as when they were placed into dry storage, including shielding and neutron absorbing materials. The used-fuel cladding and grids provide the confinement and structural rigidity relied upon to maintain the known geometric configuration and for handling and moving the fuel assembly using normal methods.

(3) *Canister integrity* means neither metal canisters nor bare casks will be breached to allow the release of helium and air ingress.

An expectation of safe dry storage of CSNF is met if sufficient confidence and basis exists that these three criteria will all remain satisfied, over a chosen duration of dry storage, and over a range of normal, off-normal, and accident storage events that are deemed credible. When all three criteria are met over extended dry storage, then the geometric and structural configuration of rods and assemblies are largely maintained, meaning the cask design to meet particular shielding, thermal, and subcriticality performance objectives are likely to remain satisfied unless other dry cask components significantly age or degrade. The standard to be met is one of credible likelihood; sufficient confidence that no credible aging process, accident condition or event exists such that any one of these criteria will not be satisfied. On the other hand, if there is a credible likelihood that any one of the three criteria will not be met, then the CSNF is considered damaged.

Similarly, an expectation of safe handling (either before or after off-site transport of CSNF) is met if sufficient confidence exists that these same three criteria will remain satisfied, during either a local transfer of a canister between casks or the unloading/reloading of fuel assemblies, and under a range of normal, off-normal, and accident handling events that are deemed as credible. If sufficient confidence is demonstrated that these conditions will be met during handling, transfer, and storage operations, together with sufficient confidence that no significant deterioration of dry-storage system components will occur, then it is assumed that the requirements of U.S. regulations as outlined below will be met.

Objectives important to storage. In the U.S., the regulations that establish the performance requirements, procedures, and criteria for the issuance of licenses regarding dry storage of light-water reactor (LWR) used fuel are provided in 10 CFR 72.[6] 10 CFR 72.236 describes the requirements for issuance of a license. Used-fuel storage systems are designed to meet four safety objectives: (1) ensure that doses from the used fuel in the storage systems are less than limits prescribed in the regulations, (2) maintain subcriticality under all credible conditions, (3) ensure that there is adequate confinement and containment of the used fuel under all credible conditions of storage and even "redundant sealing of confinement systems," and (4) allow the ready retrieval of the used fuel from the storage systems.[23] The dry-storage system "must be evaluated… to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions" [10 CFR 72.236(l)]. The Safety Analysis Reports (SAR) prepared by utilities and vendors and submitted to the NRC for licensing evaluate a given fuel storage systems' capability to meet 10 CFR 72 requirements over a 20 year[†] licensing period. At the completion of a SAR review, NRC publishes a Safety Evaluation Report (SER) that contains all of the NRC finding and conclusions from their evaluation of the SAR. Following NUREG-1536, the "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," the NRC staff procedures involve structural, thermal, confinement, shielding, and criticality reviews and evaluation.[24]

It is generally assumed that maintaining the assembly's geometric configuration may be necessary to ensure subcriticality, ready retrieval, and possibly to ensure satisfaction of the shielding requirement.[25] One aspect of retrievability is the ability to unload CSNF from a dry-storage system and repackaging it in a new container without releasing radioactive materials to the environment or causing radiation exposures in excess of the specified limits. Fuel is likely to be readily retrievable if the fuel rods and fuel assemblies remain structurally intact and unwarped. Furthermore, if the canister remains sealed with inert helium, then water (moderator) cannot enter, preventing a theoretical reactivity event. Perhaps more importantly, should helium escape and air enter, degradation of breached used fuel could begin because of oxidation, if fuel temperatures are sufficiently high.

The individual functions important to safety for a dry-storage system as established in 10 CFR 72[6] are described in NUREG-1536[24] and consist of maintaining thermal performance, radiological protection, confinement, subcriticality, and retrievability. NUREG-1536 provides different levels of priority (low, medium, or high) for which to evaluate the safety of the various performance functions. The current regulatory practice allows a very low probability (0.5 percent) of cladding failure during long-term storage.[26] NUREG/CR-6407 provides a detailed listing and description of dry-storage system components and classification categories by function.[27]

Objectives important to handling and retrievability. Regulations in 10 CFR 72 include performance requirements for handling used fuel, although storage and handling are not clearly distinguished.[6] They require that used fuel be safely and readily retrievable from storage systems for further processing or disposal and that used-fuel cladding be protected during storage against gross rupture unless the used fuel is confined (e.g., by canning, such that degradation of the used fuel would not present operational safety problems). The meaning of readily retrievable and protection against gross ruptures is not defined in the regulations, but assurance of retrievability and confinement of fuel fragments by cladding is clearly linked to the fuel rods being in an intact or undamaged condition (see NRC ISG-2,[28] "Fuel Retrievability"). What is important in the regulations and to the NRC is knowing the likelihood that some small percentage of used fuel,

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[†] NRC has under consideration a rule change that would allow for an initial 40 year license term for dry cask storage systems

initially classified and loaded into dry storage as "intact" or "undamaged," could become "damaged" during dry storage or handling.

In particular, 10 CFR 72.122 (h)(1) is focused on the safe storage and handling of used fuel after storage, and states, "The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate." 72.122 (h)(5) says CSNF, "must be packaged in a manner that allows handling and retrievability without the release of radioactive materials to the environment." CFR 72.122 (l) states, "Storage systems must be designed to allow ready retrieval of spent fuel… for further processing or disposal." CFR 72.236(m) deals with, "compatibility with removal" and handling for "ultimate disposition by the Department of Energy." The safe "handling" objective typically means reasonable assurance that CSNF "can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public" (10CFR72.3). NRC ISG-2[28] discusses criteria for the retrievability of CSNF from dry-storage systems and states the NRC view that 10 CFR 72.122(l) applies to normal and off-normal design conditions and not to accidents. For fuel to be considered retrievable, first the ability to transfer a sealed canister from a storage cask to a transportation package should be demonstrated, and, second, the ability to unload a storage cask for placement into a transportation or dual-purpose package. If fuel cannot be retrieved from the storage canister by normal means (e.g., if assemblies are in pieces and fuel rods are grossly breached), then the fuel is not retrievable "by normal means." The reason this is important for long-term storage is that not only must the fuel be safely stored at the reactor sites, but NRC regulations require that a basis be provided "to allow ready retrieval of spent fuel" without radioactive release in the distant future, if stored for an extended period. Thus, degradation mechanisms must be understood for both the fuel and the cask canister system to allow safe subsequent handling and transport.

Important fuel definitions. NRC ISG-1, "Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function," discusses various definitions related to damaged and breached fuel-rods, damaged fuel assemblies, and a suggested methodology for applying the definitions described below, but does not link the definitions directly to the storage and transport regulations.[25] The word "fuel" can refer to fuel pellets, fuel rods, and fuel assemblies. The definitions for several terms related to the physical and functional state of fuel rods and assemblies have changed over the years. For example, "damage" is not considered to be an intrinsic property of the fuel. Rather, the definitions below depend on *whether* the fuel is being stored or transported and on the particular *design* of the storage or transport system being evaluated. In other words, the definition of damaged fuel depends on the regulatory requirements applicable, which may be different for dry storage and transportation, and upon the design basis claims decided by the applicant. Here are NRC's current definitions regarding various fuel states:[25]

- **Intact fuel** – (performance classification term) a fuel rod that can perform its intended performance functions and is not breached
- **Undamaged fuel** – (performance classification term) a fuel rod that can perform its intended performance functions, but may have slight defects (hairline cracks or pinhole leaks) in the cladding

- **Damaged fuel** – (performance classification term) a fuel rod that cannot perform its intended performance functions (generally the fuel rod is grossly breached)
- **Breached fuel** – (descriptive term) any fuel-rod cladding defect that permits the release of gas from the fuel rod. A breach may be minor (such as a pinhole leak or hairline crack that releases gas or allows free entry of moisture, but not fuel particulates) or major (gross breach of cladding).
- **Grossly breached fuel** – (descriptive term), a) any breach of cladding large enough to visually see fuel pellets, b) a breach with a width greater than 1 mm, c) measurements that identify the presence of any heavy metal isotopes, or d) fuel with a minor breach but subject to active mechanism(s) that may lead to a gross breach (e.g., an oxidizing atmosphere in which fuel pellets may oxidize, unzip their cladding, and result in a gross breach). A gross breach compromises the material confinement function of a fuel rod.

Damaged fuel rods or assemblies cannot be safely stored or handled by normal means because, typically, they are grossly breached or not readily retrievable. Note that both fuel rods and fuel assemblies must be evaluated and found capable of meeting their functional requirements for storage, transportation and handling. Damaged fuel (rods or assemblies) is deemed to be in a state allowing safe storage and handling only after the damaged fuel is relocated in a "can" that ensures the necessary performance functions. At least one end of the can is an open screen to facilitate cooling and to allow the damaged fuel to be dried along with the rest of the fuel. The cans ensure that damaged fuel is retained in a configuration that can be analyzed (only very small fuel fragments could escape).

Anticipated load events and conditions during dry storage. Dry-storage cask systems must be evaluated against anticipated loading events consisting of normal, off-normal, and design-basis accident conditions including natural phenomenon events. *Normal* events are all "conditions associated with normal operation, maintenance, and testing of the ISFSI" (10 CFR 72.122(b)). The maximum level of a condition expected to occur routinely and the cask system is expected to remain fully functional with no temporary or permanent degradation of functionality or components ("accommodate the effects of" "and to withstand postulated accidents"). *Off-normal* events are the maximum level of a condition that although is not expected to occur regularly, but can be expected to occur with some moderate frequency, and for which there is a maximum level of some response or capacity. A dry cask system is expected to experience off-normal events and conditions without permanent deformation or degradation of capability to perform full design functions. NUREG 1536 recommends that the following off-normal conditions be evaluated: variations in temperatures beyond normal, failure of 10 percent of all the fuel rods combined with off-normal temperatures, failure of one of the confinement boundaries, partial blockage of air vents, human error, and equipment failure.

Accident conditions include both design-basis accidents and the design basis for events and conditions caused by natural phenomena. The dry-storage system must withstand postulated design-basis accident events and design-basis natural phenomenon events that include the specific values or ranges of values to be used for controlling parameters as reference bounds for design. NRC regulation 10 CFR 72.122(2)(i) states,

> Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunamis, and seiches, without impairing their capability to perform their intended design functions. The design bases for these structures, systems, and components must reflect: (A) Appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (B) Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena.

Of special note is a design earthquake ground motion specified for a site in accordance with 10 CFR 72.103 together with all other credible natural phenomena that may be linked or triggered by the design earthquake motion. These include volcanic activity, seismic induced floods and water waves, liquefaction potential, and slope stability threats (10 CFR 72.103). Engineered features such as berms, drainage, and shield walls that are designed and constructed to mitigate natural hazards are considered features important to safety and are part of the dry-storage system.

In-service operational limits and acceptance criteria. The Spent Fuel Project Office of NRC has issued Interim Staff Guidance 11 (ISG-11).[29] Revision 3 of ISG-11 provides "acceptance criteria needed to provide reasonable assurance that commercial used fuel is maintained in the configuration that is analyzed in the Safety Analysis Reports (SARs)" for storing CSNF, regardless of burnup level. For the purposes of used-fuel transportation, used fuel is classified into low-burnup (fuel assembly average less than or equal to 45 GWd/MTU) and high-burnup (fuel in an assembly average exceeds 45 GWd/MTU) fuel. ISG-11 also mentions that three of the four safety objectives listed earlier are common both to storage and transportation (dose limits, subcriticality, confinement) and implies the ISG-11 acceptance criteria (including the thermal criteria below) are sufficient to meet the safety objectives for handling and transport of low-burnup fuel. With respect to used-fuel handling, ISG-11 explains, "the spent fuel cladding must be protected during storage against degradation that leads to gross rupture of the fuel and must be otherwise confined such that degradation of the fuel during storage will not pose operational problems with respect to its removal from storage." According to ISG-11, the following criteria should be met to "ensure integrity of the cladding":

1. For all fuel burnups (low and high), the maximum calculated fuel cladding temperature should not exceed 400ºC (752ºF) for normal conditions of storage and short-term loading operations,
2. During loading operations, repeated thermal cycling (repeated heatup/cooldown cycles) may occur but should be limited to less than 10 cycles, with cladding temperature variations that are less than 65ºC (149ºF) each, and
3. For off-normal and accident conditions, the maximum cladding temperature should not exceed 570ºC (1058ºF).

High-burnup fuel. ISG-11 notes that the technical basis for the behavior of cladding of high-burnup (exceeding 45 GWd/MTU) used fuel during transport requires special analysis. For the transport of high-burnup fuel, ISG-11 (Rev. 3) notes *that insufficient data currently exists to provide a solid basis for general safety criteria*. As a result, approval for requests to transport high-burnup CSNF "will be handled on a case-by-case basis."

2.1 Summary

The objectives of NRC regulations for storage and handling of used nuclear fuel are aimed at ensuring that the used fuel that is stored in dry cask storage systems does not suffer damage or degradation during storage, handling or retrieval. This is largely accomplished by maintaining a helium cover gas in the canister and assuring a fixed geometric arrangement of the used fuel to provide assurance of maintaining a subcritical configuration should water enter the cask during some accident situation. Licenses have historically been granted for initial 20-year terms. NRC has recently extended the license period to a total of 60 years. Uncertainties about the behavior of high-burnup fuels for which there is sufficient data to establish criteria requires case by case reviews. Once in storage, there is no opportunity for inspection of the used fuel and canister. Monitoring is limited to inspection of cooling vents, dose readings and temperature. Physical inspections of canisters or used fuel may be required to confirm model predictions. While the opportunity of degradation of used fuel decreases with decreasing temperature, the assumption of helium presence is important to support the results of long-term storage analyses. When the NRC published the initial storage criteria, they were written for periods of 20 years; it is not clear whether the same criteria or requirements are sufficient for very long-term storage (more than 120 years) as not all degradation mechanisms have been modeled especially for high-burnup fuels. Future sections of this report will describe identified degradation mechanisms.

3. Requirements for Transportation After Dry Storage

Transportation of used fuel is subject to several federal, state, and local statutes and regulations. The two primary federal regulators are the U.S. Department of Transportation (DOT) and the NRC. DOT is responsible for regulating the safety of hazardous material shipments and NRC is responsible for licensing and regulating the receipt, possession, transfer, and use of used fuel. A 1979 memorandum of understanding[30] gives NRC the primary responsibility, in consultation with DOT, for the development of standards and regulations for the design, performance, and inspection of transportation packages for used fuel. Accordingly, NRC establishes design standards for transport casks; reviews and certifies cask designs before their use; and conducts inspections to ensure casks are designed, fabricated, and used, and maintained in accordance with regulations. The NRC also has the primary responsibility for providing physical protection requirements for used-fuel shipments. DOT has the primary responsibility, in consultation with NRC, for establishing regulations for shippers and carriers regarding radiological controls, hazard communication (labeling packages and placarding vehicles); equipment maintenance requirements; carrier personnel qualifications; and procedures for loading, unloading, handling, and storage during transport. In addition, DOT has the primary responsibility for special transport controls for radiation safety during transport including all inspection transportation activities of carriers. Highway shipments of used fuel are also subject to state inspection and state permitting.

While dry storage of used fuel does not normally impose any new stresses on used fuel or storage systems, the canisters or casks must, at some point, be transported from the reactor sites. The NRC approach for licensing transportation of used nuclear fuel requires a complete description of the transport package and all its contents according to Title 10, Part 71 of the Code of Federal Regulations (10 CFR 71).[31] Given an adequate initial description, a package is evaluated according to the performance requirements in 10 CFR 71, which are listed below in subsection 3.1. In addition to the 10 CFR 71 regulation, other NRC regulatory requirements specify more detailed performance requirements for fuel rods and fuel assemblies that are transported. Following NUREG-1617, the "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," the NRC staff carry out detailed structural, thermal, containment, shielding, and criticality reviews and evaluations of submitted applications. This includes demonstration of compliance after subjecting the package to normal transport operations and a series of severe, but credible hypothetical accident conditions. So after extended dry storage, there must be sufficient technical basis to fully describe the state of the fuel and confirmatory calculations to ensure adequate package radiation containment, shielding, thermal performance, and subcriticality.

Depending on the condition and degradation of the used fuel after extended storage in the canister or cask systems, some additional fuel failures may occur during transportation. The transportation regulations are largely focused on the integrity of the casks during transit and maintaining subcriticality of the used fuel. 10 CFR 71 does not expressly require that fuel rods be intact or undamaged, and there is provision made to allow for specially designed "cans" to hold damaged fuel. However, if fuel rods are initially assumed to be intact when placed into the transport cask, the current evaluation criteria of NUREG-1617 do not appear to allow for *many* fuel rods to rupture, during transportation mechanical analyses, and still permit the transport package to be licensed. The initial description, classification and geometry of fuel are used to assess compliance with the performance requirements listed in NUREG-1617. Currently, for a utility to obtain a certificate of compliance for a transport package, NRC review staff require that, generally, fuel classified as undamaged when loaded into the package maintain their classification during normal transport operations or under hypothetical accident conditions. This is to ensure that the transported fuel assemblies will be retrievable at every type of facility they may be shipped to. This means the retrievability requirement of 10 CFR 72 is, in practice, added as a transportation requirement.

However, the NRC staff may allow a small number of undamaged fuel rods to become damaged. Thus, the minimum requirements necessary to obtain a transportation certificate of compliance after extended storage of used fuel is not clear at present. What is important to the NRC review staff is knowing the number and location of used-fuel rods and assemblies initially classified and loaded into a transport packages as “intact” or “undamaged,” and the number and location of these same rods and assemblies that are likely to become “damaged” during normal transport or under hypothetical accident conditions. Again, it is not clear how many fuel rods, or perhaps how many in what location, regulators will allow to become damaged in a submitted analysis. In any event, it can be concluded that it is important to understand the behavior of both fuel cladding and the structural elements of assemblies *after extended dry storage*, so that the required fuel and fuel-assembly mechanical modeling can be performed with adequate confidence in the predicted results.

3.1 Transportation Package Regulations

The regulations that establish the safety standards, requirements, procedures, and criteria for the issuance of licenses regarding packaging and transport of packages (casks) of LWR used fuel are provided in NRC regulations (10 CFR 71).[31] A package holding used-fuel is classified as a “Type B” material package from a regulations standpoint, which means a particular subset of requirements apply within 10 CFR 71. Of key interest is package performance: the ability of a transportation package to maintain a high level of containment effectiveness under extreme mechanical forces (impact, crash and penetration), thermal loading, and water immersion conditions that represent the forces that may be experienced during a severe accident. Accordingly, regulations and criteria are specified that cover normal conditions of transport and hypothetical accident conditions (the sequence of six accident conditions are described in 10 CFR 71.73). The NRC amended 10 CFR 71 in 2004 to reflect the International Atomic Energy Agency (IAEA) standards for the safe transportation of radioactive materials. The requirements for certification (10 CFR 71.73) are derived directly from IAEA standards.

Transportation package regulations are intended so that approved packages can be safely transported and not endanger emergency responders or the general public. The NRC application and review process to certify a package for transport is technically rigorous. The four key transport objectives addressed in regulations are: containment, shielding, criticality safety, and heat management. Specific to 10 CFR 71, five basic types of performance requirements and one characterization requirement must be met. During “normal conditions of transport,” which includes minor mishaps because of rough handling or exposure to weather), a more strict set of requirements must be satisfied than is required for the sequence of “hypothetical accident conditions” defined in subsection 3.1.1. According to current NRC staff guidance, after simulation of the fuel-cladding behavior in cask performance tests under the specified "hypothetical accident conditions, the licensee must assure that there is no significant cladding failure."[29] These are the specific regulatory requirements:

1. Limit dose rates on external package surfaces to acceptable levels (10 CFR 71.47 and 10 CFR 71.51).[32] This specifies the contact dose on the surface of the package not exceed 200 mrem/hr for normal transport.
2. Prevent or limit the release of radioactive contents with the requirements for normal transport being more restrictive than those for hypothetical accident conditions (10 CFR 71.51).[33]
3. Prevent an unsafe configuration (i.e., accidental criticality) of used fuel for both normal and accident conditions (10 CFR 71.55).[34]
4. Preclude any "substantial reduction in the effectiveness of the packaging" for normal transport conditions including limits on the reduction of effective package volume and fuel spacing (10 CFR 71.43 and 10 CFR 71.55).[35]
5. Limit temperatures on external package surfaces to specified levels (10 CFR 71.43) for both normal and accident conditions.[36] This temperature should not exceed 122 F under ordinary circumstances.
6. A description of the proposed package contents, including "chemical and physical form" initially, and of any expected variations during normal transport and after the hypothetical accident condition tests (10 CFR 71.33(b) and 10 CFR 71.55).[37]

As mentioned previously in Section 3 of this report, ISG-11 Rev. 3 indicates that for the transport of high-burnup fuel, insufficient data currently exists to provide general safety criteria in accordance with the regulations noted above. As a result, approval for requests to transport high-burnup CSNF "will be handled on a case-by-case basis using the criteria given in 10 CFR 71.55, 10 CFR 71.43(f), and 10 CFR 71.51."

Several different truck and rail package designs have been developed to meet these performance specifications. Most packages consist of a hollow cylindrical body that is open at one end. The body is typically constructed of multiple layers of the following material: steel is used to provide structural strength and durability; steel, lead, depleted uranium, or concrete is used to provide shielding against gamma radiation; and borated polymers, or concrete is used to provide shielding against neutrons. The package closure system consists of one or two steel lids with elastomer or metal seals that are attached to the open end of the package body with steel bolts. Packages are typically designed with impact limiters to absorb mechanical forces generated in the event of accidents and to provide thermal protection in case of fires. The impact limiters, generally attached to the package ends, typically use wood, rigid foam, or honeycombed metal as the energy absorber material. Some packages have metal fins on the exterior surfaces of the package body for additional impact protection and heat dissipation. The design of the interior of a transportation package is similar to a storage cask.

3.1.1 Package Performance Testing

As indentified in 10 CFR 71, shipping casks or packages are required to survive a sequence of tests simulating "normal conditions of transport" (10 CFR 71.71) and "hypothetical accident conditions" (10 CFR 71.73). The more rigorous suite of accident tests are to be performed in this sequence:

- *free-drop test*: drop package 9 m (30 ft) onto an essentially unyielding horizontal surface (strike speed will be about 48 kph or 30 mph) with the package orientation arranged so that its weakest point is struck,
- *puncture test*: drop package 1 m onto a vertical, 15 cm (6 in) diameter, cylindrical mild steel bar striking the package at its most vulnerable spot,
- *thermal test*: exposure of packaged, fully engulfed in a fire of at least 800ºC (1470ºF) for a period of 30 minutes with less than the specified loss of containment effectiveness, and
- *immersion test*: package placed under a head of water of at least 15 m (50 ft) or an external pressure of about 150 kPa (21.7 psi or 3130 psf) for at least 8 hours.

The NRC permits quantitative analysis (e.g., computer simulations using physics based finite element or finite difference models), scale-model (typically one-quarter or one-half scale), and full-scale testing of packages or package components, and comparisons with existing approved package designs to be used to demonstrate compliance with the regulations. Scale model testing is limited in that some of the most important weak points in a complex structure often may not scale well, for instance bolts and welds.[38] These situations require special analyses. Testing of full-scale actual packages (including both *certification* tests under actual accident conditions and *demonstration* tests under simulated accident conditions) is not a requirement of the regulations. The NRC has certified several transportation package designs that include dual and multi-purpose packages designed for both transportation and storage.

In addition, the federal regulations require design, manufacturing, and testing activities to be carried out using an approved quality assurance program. Packages are not required to be designed and tested according to quantitative risk assessment standards, but rather to standards including conservative assumptions and "built-in" design margins for material properties. Furthermore, the regulations "were not derived from a comprehensive bounding analysis of all possible extreme thermomechanical conditions resulting from package mishandling and accidents. Rather, their development… was based on then-available [1960s] data on typical impacts… and thermal environments.… As such, they do not necessarily reflect the most extreme conditions that might be encountered during spent fuel" shipments.[39] For example, the impact velocity and accelerations of a package in the free-fall drop test (against an unyielding surface) may be lower than extreme real world crashes. Although it should be noted that in earlier tests of rail crash tests, the train impact on a rail cask was over 80 mph.

Currently, if used fuel is stored in a dual-purpose storage system, transportation certification requires that the applicant show that the stored fuel and container is safe for transport. Given that fuel may be stored for decades before it is transported, and the possibility of degradation of fuel and corrosive deterioration of canisters over this time, applicants for transportation certificates will need to rigorously analyze such "aging" problems, which they have not needed to do in the past. It is possible that either the dual-purpose canister or the transportation overpack will have aged over its life. If so, the former numerical analysis and scale modeling of such transport packages may not reflect the actual behavior of aged fuel and packages.

3.1.2 Package Performance Modeling Studies

Because of the expense involved with some performance testing, computer simulation modeling is routinely used to predict the thermomechanical behaviors of truck and rail packages under a range of extreme loading conditions. "Generic" packages are used and loading conditions applied are derived from historical accident reconstructions. Partial verification of the accident simulation results can be found by comparing predicted behavior with that from scale-model tests. Full-scale or quarter-scale testing has not been carried out in the U.S. for several decades (although Germany has recently carried out such tests). However, it is known that typical structural weak points (bolts, welds, and adhesives) and in situ stress conditions may not scale well. Existing cracks and other initiating conditions also do not scale well.[40] With the development of sophisticated analytical methods and high-speed computers, the need for full-scale testing in this and other research areas has diminished, given the quality and validation capability of scale-model testing.

Physical tests (which could be either full or reduced scale) are needed to validate modeling and analysis of aged and extended-burnup fuel performance under both normal and severe accident conditions of transport. Such validation is useful in building a convincing argument that any given package can transport fuel safely in support of the transport-cask design-review process.

3.2 Summary

The NRC transportation requirements, as described above, appear to have been written for transportation of CSNF after a relatively short storage period because degradation of fuel rods and fuel assemblies is not clearly accounted for. Meeting the current specified packaging requirements after a period of extended dry storage involves satisfaction of four objectives of safe radioactive material transport: containment, shielding, criticality safety, and heat management.. The focus of the regulations in 10 CFR 71 is largely on the integrity of the shipping cask and avoiding criticality, and not necessarily on the condition of the used fuel inside, provided certain performance requirements are met. However, current certification practice of transport packages requires applications to show that most fuel rods cannot be classified as damaged after transport. Consequently, it is important that at the time of transport, and after extended dry storage, (1) the initial condition and mechanical properties of the fuel and fuel-assembly structural elements are sufficiently characterized and (2) the behavior of fuel rods and fuel assemblies during normal and accident transport conditions can be sufficiently modeled to know whether transport requirements will be met.

The current NRC practice of requiring retrievability for both used fuel storage and transport imply that the existing storage and transportation requirements should be integrated and consideration be given, possibly, to the transport destination since the facility requirements may be quite different. Other regulations may govern the receiving condition and requirements for handling at the receiving location be that a repository, reprocessing plant or other interim storage facility.

Should the used fuel ultimately need to be removed from its storage system or its dual-use cask for transport repackaging, it is even more important to understand the condition of the used fuel before opening of the primary containment vessel. For low and high-burnup fuels, spot-checking on the condition of the used fuel over regular time periods on a representative-sampling basis should be done to confirm and validate existing degradation models to provide confidence in the modeling assessments.

The NRC will need to clarify existing transportation requirements consistent with the objectives of extended storage and, possibly, the transport destination since the requirements may be quite different if the used fuel needs to be handled or directly disposed of in the canister system.

4. Significant Dry-Storage Features, Conditions, Events, and Processes

4.1. System Features and Components

4.1.1. State of the Used Fuel-Pellets Before Dry Storage

Fresh commercial nuclear fuel consists of solid ceramic pellets of uranium (or in the future, some mixed uranium-plutonium) oxide contained in zirconium-alloy fuel cladding tubes. The pellets are about 0.4 to 0.65 in. (1.0 to 1.65 cm) and about 0.3 to 0.5 in. (0.8 to 1.25 cm) in diameter. A fuel rod (also referred to as a fuel pin) is highly corrosion resistant to withstand protracted operation in high-temperature water in the intense radiation environment of a reactor core.

Reactor fuel burnup is a measure of the energy that has been produced by the fuel. High-burnup fuels generally have had longer in-reactor residence times than low-burnup fuels (fuel in an assembly averages less than or equal to 45 GWd/MTU). For light-water reactors (LWRs), the "current fuel design has reached its limit at a burnup estimated to be about 80 [GWd/MTU]."[41] The current operating reactor burnup limit in the U.S. "is 62 GWd/MTU (peak rod), and the nuclear industry seeks to increase peak rod burnup from 62 to 75 GWd/MTU."[42] Generally, high-burnup fuel has remained in wet storage because dry-storage casks were not licensed for fuels with higher burnup than 50 GWd/MTU until recently (Tables 1 and 2). Some older casks (e.g. TN-40), licensed for 45 GWd/MTU are being redesigned for use with high-burnup fuels (e.g., TN-40HT for 60 GWd/MTU).[43]

Used commercial nuclear-fuel pellets consist of a mixture of uranium oxide, various fission products, and other metals including transuranic elements. Fresh UO_2 pellets contain typically about 3 to 5 percent by weight ^{235}U, which sustains the fission chain reaction in a nuclear reactor, and about 95 to 97 percent ^{238}U, which can capture neutrons to eventually produce plutonium and other heavy elements (known as actinides). In addition, the fission and neutron-capture reactions convert uranium, plutonium, and other actinides into nearly 300 other radionuclides in the fuel (constituting approximately 5-7 percent of the used-fuel mass), including fission products such as strontium-90 (^{90}Sr), cesium-137 (^{137}Cs), krypton-85 (^{85}Kr), iodine-129 (^{129}I), and technetium-99 (^{99}Tc) and actinides such as neptunium-237 (^{237}Np).

The density of the fuel pellets evolves in the reactor because of formation of solid fission products and fission gases, mostly krypton (Kr) and xenon (Xe), which cause pellet swelling along with accumulation of induced stresses. Steep thermal gradients that develop in a pellet under irradiation (because of the low thermal conductivity of the pellet) induce significant internal stresses (that exceed the fracture strength of UO_2) and non-uniform changes in density. As a consequence, fracturing and fragmentation of fuel pellets occur during irradiation (Figure 14), which increase the pellet effective volume further.[44,45]

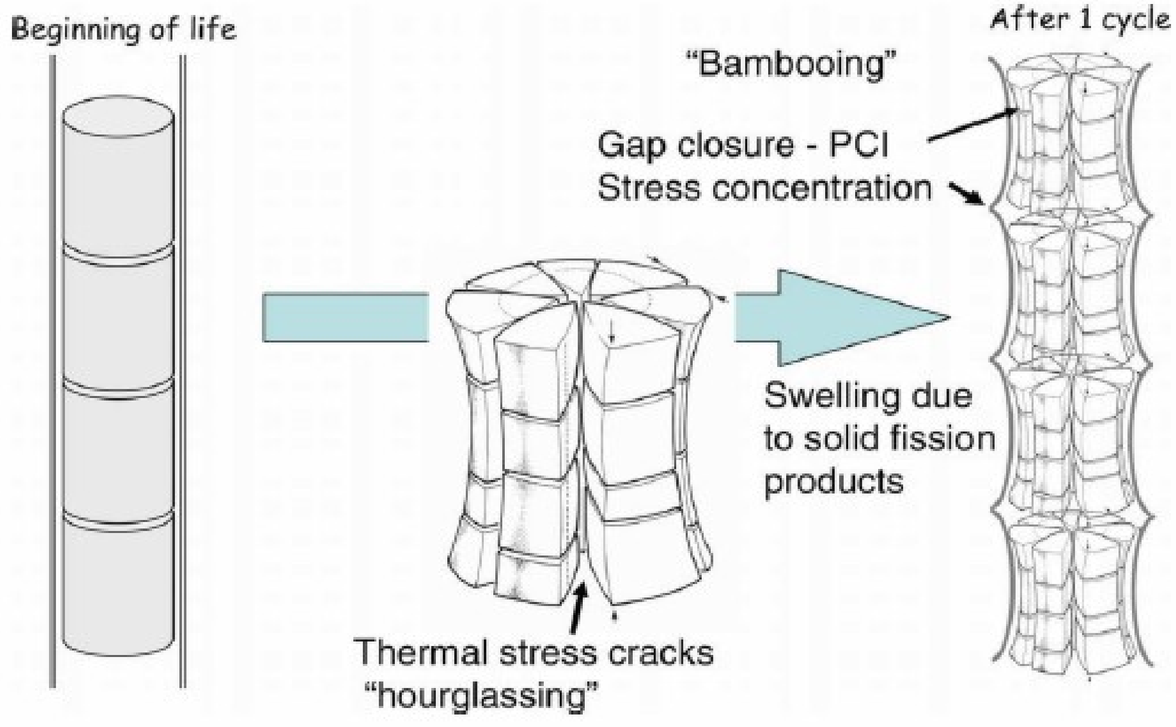


Figure 14. Effects of Fuel-Pellet Swelling and Thermal Stresses[46]
(Note: PCI stands for pellet-cladding interaction)

Used fuel has a heterogeneous physical structure of grains, grain boundaries, a porous rim structure, and cracks (see Figure 15). In addition, the chemical composition of used fuel-pellets is heterogeneous (see Figure 16), including fission product gases, typically Xe and Kr, metallic fission products, oxide precipitates, and others as explained by Bruno and Ewing.[5] In-reactor, the pellet temperature can be as high as 1,700°C at the center of the pellet, decreasing to about 400°C (the regulatory guidance limit) at its rim or outer boundary. Normal reactor operation causes fuel-pellet temperatures to vary significantly over time and leads to extensive pellet fracturing into 10 to 30 pieces with some additional powder creation. Irradiation-induced temperatures and gaseous diffusion leads to nonhomogeneous distribution of fission products within a pellet [see Figure 15]. In addition, the degree of fuel burnup is non-uniform within a pellet. "Higher burn-ups at the edge of the pellet lead to higher concentrations of ^{239}Pu at the fuel edge, an increase in porosity, and polygonization of the UO_2 grains resulting in a reduction in the size of individual grains (~0.15 to 0.3 µm) – the so-called 'rim effect'."[5]

Failed-Fuel Oxidation.[47] Fuel pellets in rods that fail (breached cladding) during reactor service may be subjected to thermal and steam oxidation. The introduction of water complicates the oxidation process by facilitating formation of hydrated oxidation products. The presence of variable moisture content (and especially of relative humidity above 50 percent) may have a significant effect on oxidation of CSNF, but more research is needed to confirm and quantify this effect. A survey of oxidation research shows that the oxidation rates for irradiated fuels are significantly higher than for unirradiated UO_2.[47]

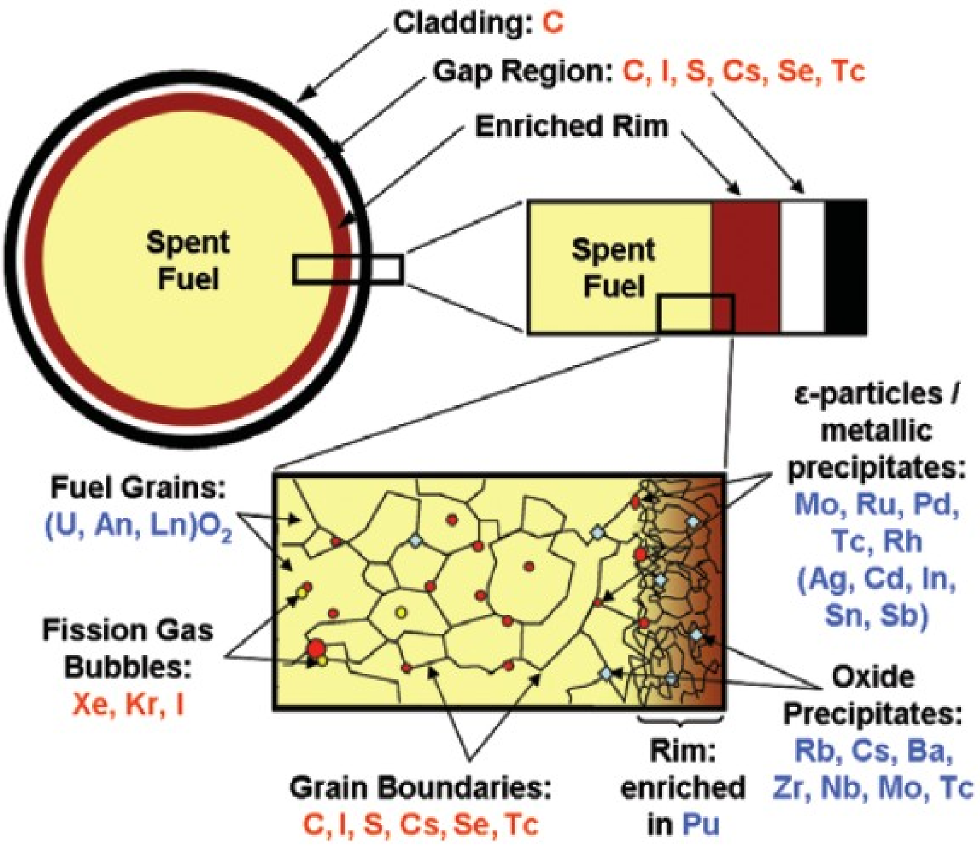


Figure 15. Schematic Illustration of the Microstructure of Used Fuel-Pellet and the Distribution of Actinides and Fission Products Following Burnup in a Reactor[5]

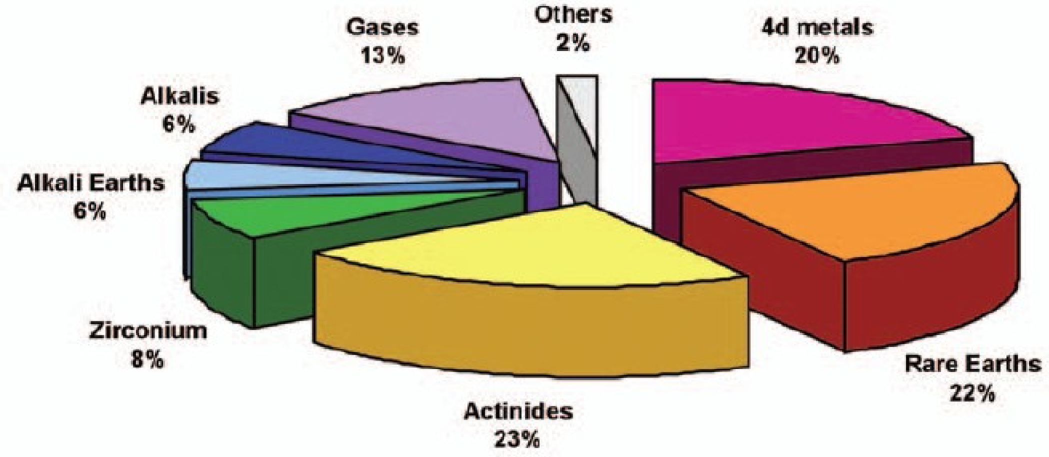


Figure 16. Pie Diagram Showing the Relative Proportions of the Major Fission Product and Transuranium Elements Found in Used Fuel-Pellets at Moderate Burnup[5]

(Note: “4d metals” corresponds to the filling of the 4d orbital; it starts from ytterbium whose atomic number is 39 and includes 10 elements till cadmium whose atomic number is 48: Y, Zr, Nb, Mo, Tc, Ru, Rh, Pd, Ag, and Cd)

High-Burnup Fuel Characteristics.[42] As defined earlier, high-burnup fuel has an average burnup that exceeds 45 GWd/MTU. Closer investigation reveals that above about 40 GWd/MTU, the surface microstructure of the UO_2 fuel-pellet begins to change. Burnup does not occur uniformly across a fuel pellet cross-section. Generally, burnup remains fairly uniform from the center of the pellet outward to near the pellet edge at which point it spikes up rapidly. For example, for the cross-sections averaged burnup of 44.3 GWd/MTU in one pellet, burnup remained around 40 GWd/MTU and spiked to 70 at the edge (this would be considered low-burnup fuel). The high-burnup rim zone is characterized by a much smaller grain size, about double the porosity, and larger pore size. The differences become more significant as burnup increases. The net result is that the rim material has high fracture toughness, which makes it resistant to breaking down into fine particles. Only a few percent of total fission gases contained by the pellet matrix material are released into the fuel-rod void space; most are retained in the porous rim.

4.1.2. State of the Fuel Cladding Before Dry Storage

Figure 17 shows the geometry of a typical LWR generic fuel-rod. It consists of a cladding tube typically 11.5 to 14.75 ft. (3.5 to 4.5 m) long and filled with a stack of fuel pellets about 3.5 m high. The open space in the tube above the fuel pellets is called the plenum and is designed to hold fission gases released from the fuel without overpressurizing the cladding. Typical tube nominal outside diameter is 12 mm for BWR rods and 8 mm for PWR rods with a nominal wall thickness of 0.6 mm. When fabricated the cladding is designed with a small gap (about 0.1 mm) between the external radius of the fuel pellets and the internal radius of the tube; this gap closes during irradation. Fuel rods are bundled together, with a 0.12 to 0.18 in. (0.3 to 0.45 cm) space left between each to form a square assembly measuring about 6 to 9 in. (15 to 23 cm) on a side. The number of fuel rods in a fuel assembly range from 62 to 96 for BWRs and from 164 to 264 fuel rods for PWRs depending on vendor design and when in use. Fuel assemblies consist of the fuel rods, spacer fuel-grids, and control rods. The fuel grids consist of an arrangement of interlocked metal straps that maintain the lateral spacing between the fuel rods. Grids typically have spring fingers and dimples that grip and support the fuel rods. Shown on Figure 18 are typical PWR and BWR fuel assemblies. The control rod is shown inserted into the PWR assembly and between the BWR assemblies (courtesy Westinghouse and General Electric). Each fuel assembly contains the fuel rods.

Cladding functions. Fuel cladding is designed to ensure that the used fuel remains in a known geometric configuration during storage and transportation and that inert He gas is maintained around the fuel pellets. The fuel rod is designed to act as a pressure vessel and contains the fuel-rod inert-gas pressurization. Fuel rods are pressurized internally with helium to decrease clad creepdown during in-reactor operation. Total rod internal pressure is a function of the fill helium, the fission gas release during service, and gas temperature. The typical helium density ranges from 0.7 to 0.8 grams/liter. The historical original helium fill gas pressure of LWR fuel rods is "between 1.5 to 3.5 MPa (200 to 500 psi) at 20ºC for PWRs and between 0.3 to 0.7 MPa (50 to 120 psi) at 20ºC for BWRs."[48] End-of-life (start of dry storage) rod gas-pressure estimates made in 1977 at 25ºC was 1.7 to 3.8 MPa for PWR fuel and 1.4 to 2.0 MPa for BWR fuel rods.[49] The increase in gas pressures is because of the fission-gas generation rate. the Electric Power Research Institute (EPRI) in 2007 combined end-of-life PWR rod pressure data from several sources and derived "representative" values ranging from 3.0 to 4.5 MPa.[50] The EPRI report also stated that "higher end-of-life pressures, by a factor of 2, can be found in fuel rods containing boron-coated pellets." Beside consideration of fuel pellets, cladding represents the first physical barrier against release of radioactive material into the environment.

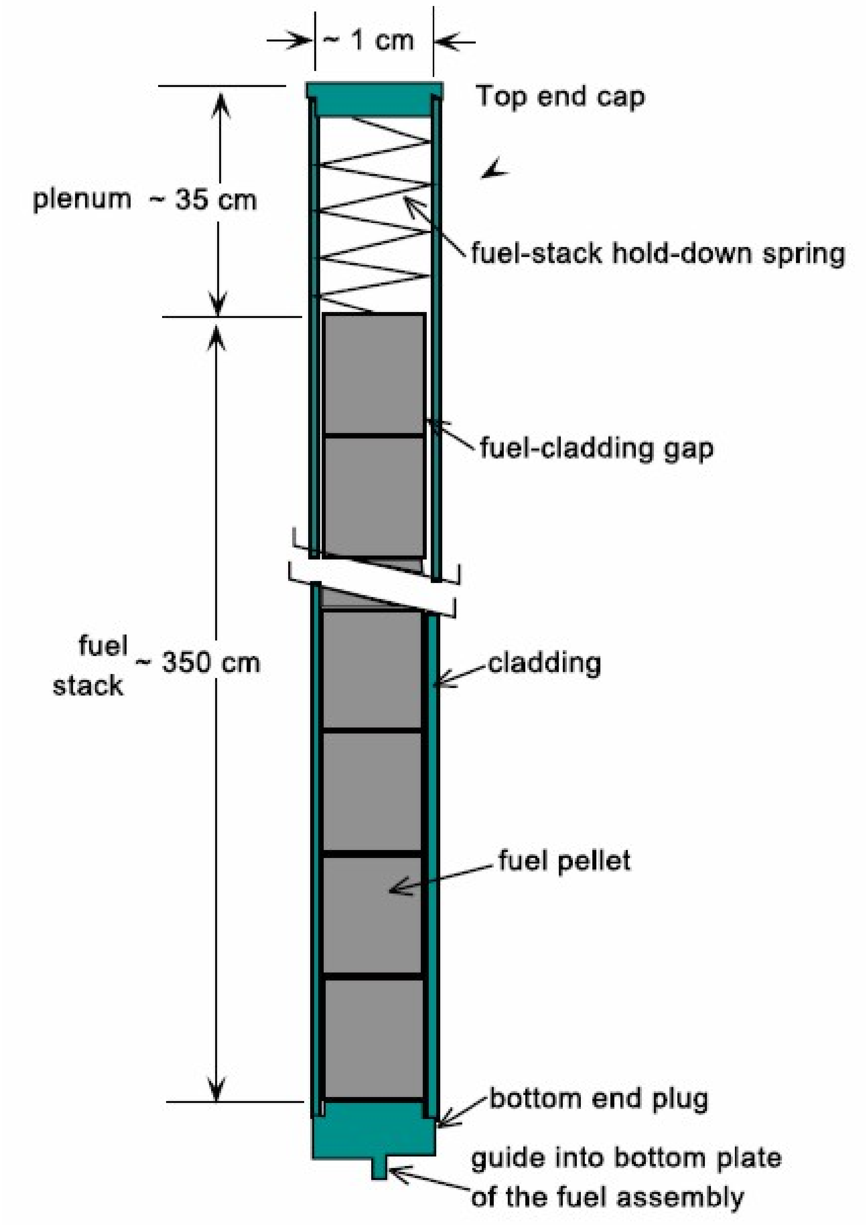


Figure 17. A Generic LWR Fuel Rod[51]

Figure 18. Typical PWR and BWR Fuel Assemblies

Used with permission of Westinghouse (left)

http://gepower.com/prod_serv/products/nuclear_energy/en/downloads/gnf2_adv_poster.pdf (right)

Cladding alloy materials. Cladding mechanical properties are firstly dependent on the actual elemental composition of the alloys used. Originally, stainless steel was used for cladding. One BWR (La Crosse) and four PWR reactors (Yankee Rowe, Connecticut Yankee, Indian Point Unit 1, and San Onofre Unit 1) loaded a total of 2,179 stainless steel fuel assemblies for a total of about 400,000 fuel rods, which represents about 1 percent of all commercial fuel rods.[52] The traditional zirconium alloy for U.S. PWR cladding tubes is Zircaloy-4 with approximately 1.5 weight-percent tin as the major alloying element. Most BWR fuel cladding of recent decades was made of Zircaloy-2 alloy. The vast majority of laboratory test data relies on Zircaloy-4 data. Today new types of cladding alloys are being commonly used (discussed later), designed for high-burnup fuels.

Second in importance in determining behavioral cladding properties is the "microstructure" meaning the metallurgical details of the alloy including: grain size and shape, dislocation density, alloy phase presence and distribution, "solute" composition, and texture. For a given alloy composition, these microstructure components are designed and fixed during fabrication by the processes such as cold and hot working, quenching and annealing.

A typical BWR Zircaloy-2 fuel rod initially has the following microstructure characteristics: equiaxed grains with an average diameter (size) in the 4-8μm range, a very-low dislocation density, a completely alpha-phase matrix, second phase precipitates of crystalline form uniformly distributed, of type $Zr(Fe,Cr)_2$ and $Zr_2(Fe,Ni)$ chosen for corrosion performance, and a matrix "solute" content of about 13,000 ppm Sn and 1400 ppm O added for strength considerations. For a typical PWR cold-worked/stress-relieved Zircaloy-4 rod, the following microstructure differences are most prominent: the grain size is highly elongated in the axial direction, the dislocation density is very high because of the cold working, the second phase precipitates are only of type $Zr(Fe,Cr)_2$ due to the alloy chemical composition and are roughly twice the size of Zr-2 precipitates to optimize the PWR uniform corrosion resistance.[53]

Because of the more severe conditions created by burning fuel to higher levels, new cladding materials have been developed for in-reactor service and employed by vendors such as Areva's M5 alloy, Westinghouse's optimized ZIRLO, Siemen's Duplex, and Mitsubishi's M-MDA material. Currently there is much more behavioral data available on Zircaloy-2 and -4 cladding, but work is ongoing to study the new cladding materials (mostly proprietary). From the limited information reviewed it appears new cladding research is focused primarily on in-reactor behavior and not behavior during extended storage.

Cladding defects. Some rods can have small manufacturing micro-defects or handling induced defects including small partial-depth cladding wall cracks, weld defects, moisture or organics contamination of cladding or pellets leading to hydriding, and excessive gap at spacers leading to vibration and fretting. It is rare that such defects lead to the fuel being classified as damaged for in-reactor use or during dry storage, unless the defect is systematic in nature. Systematic defects are material design flaws that, when discovered, are corrected in future designs. Whereas used-fuel design flaws for reactor performance can be discovered by studying fuel failures in-reactor, there is a small chance that design flaws for very long-term dry-storage and transport performance will not be discovered for decades or longer.

4.1.2.1. In-Reactor

Over the years that fuel rods and assemblies are manufactured, placed into use, and removed from the reactor, they undergo many changes that can alter their physical structure and mechanical properties. Some of these conditions will be described in this section, including changing gas pressures, irradiation, interaction with fuel pellets, cladding oxidation and creep, hydride formation, and aspects of reactor service. All of these circumstances lead to physical damage of fuel rods and assemblies, but as described in Section 2, by the regulatory definition, the fuel may still be considered “undamaged” if it retains its ability to perform its intended design function.

One type of damage that occurs in both Zircaloy-2 and Zircaloy-4 cladding material in a reactor is irradiation growth.[54,55] During irradiation, zirconium alloys experience a change in shape at constant volume in the absence of stress: fuel rods increase along their length and shrink in diameter. After one year of irradiation, the growth strain ranges from about 0.1 to 2.0 percent, with slightly larger negative strains in the radial direction. Under typical reactor conditions, the growth strain may be about 0.2 percent, increasing a 3-m long fuel rod about 6 mm; the reduction in diameter due to this effect is small. Irradiation growth occurs because irradiation causes dislocations in the anisotropic crystal-zirconium-alloy structure, which causes irrecoverable deformation anisotropy. The effect of irradiation growth damage on subsequent damage mechanisms affecting cladding during dry storage is not well understood.

During normal reactor service, the plenum gas-pressure increases over the initial He fill gas-pressure because of the fuel-pellet release of He, creation of fission-gas products Kr and Xe, and rising temperatures. Release of fission gas into the fuel-rod gas space occurs by diffusion (where the diffusion coefficient varies with temperature), cracking of the pellets, and fission fragments causing knockout and direct recoil.[56] The first consequence is that in vertically oriented fuel rods, the heavier Kr and Xe gases displace He gas upward, which results in a greatly decreased thermal conductivity lower in the rod, unless the pellet-cladding gap is closed by fuel swelling. If the gap is not closed, the fuel temperature rises in this zone, leading to increased fission-gas release.[51] The second significant consequence is if sufficient fission-product gases increase the gas pressure within the cladding to a higher level than the external cladding pressures in the reactor. If so, then the cladding expands (and may start to creep outwards) and “lifts off” the fuel pellets, increasing the gap and the in situ cladding tensile-stresses. A larger gap may result in an increase in the fuel-pellet temperature that can feed a positive thermal feedback loop, increasing the fission gas product release rate due to fuel expansion, increasing fuel-rod overpressure even more, and leading to more outward creep and an even larger gap.[51,53]

In addition, cladding helps contain fuel-pellet swelling if the radial gap between the pellet and the cladding (plenum gap) entirely closes. During burnup in a reactor, three mechanisms contribute roughly equally to decrease the cladding: fuel thermal expansion, cladding creepdown, and pellet-fragment relocation.[57] In-reactor operating conditions impose an external hydrostatic force on the cladding, causing it to creep down over time into eventual contact with the fuel. Zircaloy cladding deforms non-uniformly (in an oval shape) under these conditions. Cladding creepdown varies primarily as a function of hoop stress (the tensile stress in the cladding wall in the circumferential direction), temperature, and exposure time.[58] Further cladding creepdown is not an issue once used fuel rods have been removed from the reactor because external water pressures on the rods in the pool or due to the backfill helium gas pressure in a dry-storage canister are minimal.

Consequently, the gap size is dependent upon the initial size of gap, the time in reactor (burnup), and the temperature profile of both the fuel pellets and cladding. However, there is no reliable analytical approach to predict the fuel/clad gap for a given fuel rod.[47]

In-reactor, cracks can be generated within the cladding because of this fuel-pellet against cladding thermal-mechanical interaction. These "mini" or incipient cracks can be observed in an irregular pattern located on the inner surface of fuel cladding.[59,60] The cladding tube wall is forced to conform to the "hourglass" shape of pellets giving a bamboo shape similar to Figure 14 (deformation is exaggerated). As a result, the cladding is stressed in tension, both radially and axially.[51] The importance of the existence of part-wall mini cracks in intact rods is that they may be precursors for three other damage mechanisms that will be discussed later: creep rupture, stress corrosion cracking and delayed hydride cracking.

Internal and external corrosion of the cladding (Figure 19) can occur while in-reactor. Where the inner surface of the cladding is in contact with the fuel pellets, the cladding oxidizes "non-uniformly in depths of the order of some micrometers" by reduction of the UO_2 pellets.[61] The outer surface of the cladding oxidizes because of high-temperature PWR water chemistry (the Zr metal corrodes in a chemical reaction). This is a problem because oxidization of cladding weakens its mechanical properties, it effectively reduces the Zr-metal cladding tube cross-section, and it decreases the cladding thermal conductivity (due to the zirconium-oxide product). In BWR reactors, water temperatures are about 20ºC hotter than PWRs. If cracks in the uniform zirconium-oxide film occur, water can react with the metal cladding and form white ZrO_2 nodules (left-hand photograph in Figure 19). In PWRs more-uniform corrosion (right-hand photograph) can occur along with deposition of Fe, Ni and Cr CRUD (Chalk River Unidentified Deposits, all corrosion products of stainless steel piping).[62] The presence of crud can induce localized corrosion on the cladding. The thickness of the oxide layer formed on the external surface of the cladding is typically greater than the thickness of metal oxidized to form the oxide layer. Figure 20 shows the cladding outer-surface oxide-layer thickness, varying with average burnup, of more than 4,400 measurements from commercial fuel rods irradiated in reactors from around the world.[63] It can be seen that at a burnup of less than about 40 GWd/MTU, the external oxide thickness is less than 30 microns with relatively low statistical variation. Above an average burnup of 40 GWd/MTU the average external oxide-thickness increases 2-3 fold with much greater statistical variation. This change of physical behavior starts to occur at a burnup value that is close to 45 GWd/MTU, the conventional value used to separate low and high-burnup fuels. For high burn-up and extended fuel-cycle operation of PWRs, aqueous corrosion of Zr-fuel cladding alloys is a major limitation.[64] The unoxidized metal in the cladding walls of high-burnup fuel (i.e., fuel with average burnups exceeding 45 GWd/MTU) may have become relatively thin from in-reactor formation of oxides or zirconium hydride.[65]

In-reactor, a loss-of-coolant-accident results in high fuel temperatures. When emergency coolant is introduced back to the core, the fuel rods are quenched during rewetting, which may result in large thermal-shock stresses. An oxidation limit criterion was developed based on the retention of Zircaloy cladding ductility (no breach) at the reflooding temperature. 10 CFR 50 defines the maximum cladding oxidation limit to be that nowhere on a fuel rod can the oxidation thickness be more than 17 percent of the original thickness.[66] Accordingly, the metal cladding thickness of CSNF rods placed into dry storage can be assumed to be at least 83 percent of their original manufactured thickness.

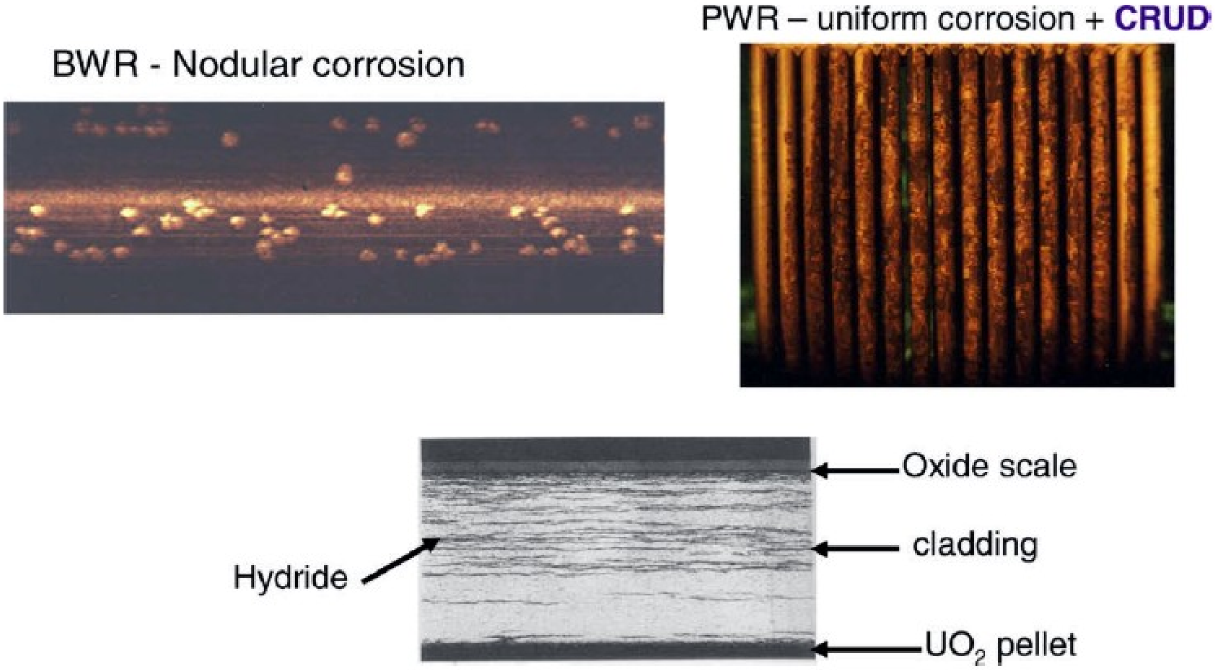


Figure 19. Corrosion of Zirconium-Alloy LWR Cladding[62]

A second consequence of in-reactor corrosion, particularly in PWRs, is partial absorption of the corrosion-product hydrogen into the Zircaloy metal (hydrogen pickup) that can lead to hydrogen embrittlement (loss of cladding ductility) of the cladding. When the concentration of hydrogen in solution in the Zircaloy matrix exceeds the solubility limit, the excess hydrogen precipitates as hydrides. Cladding is manufactured with a texture to ensure that hydrides precipitate generally in a circumferential direction upon reactor cooldown (lower photograph in Figure 19). Hydride precipitation can be evenly distributed or dominate in certain areas.[67] Both cladding hydrogen-content and effective wall-thickness are correlated to the amount of oxidation that occurs on the outer surface of the cladding. Plotting more than 4,400 measurements from commercial fuel-rods taken from reactors around the world, Figure 20 shows the maximum outer-surface oxide-layer thickness data in low-Sn Zircaloy-4 cladding plotted as a function of burnup.[63]

Taking these oxide thickness measurements, the maximum wall thickness average (MWTA) hydrogen content can be calculated using a hydrogen evolution model. Figure 21 plots the wall-average hydrogen content in low-Sn Zircaloy-4 cladding as a function of burnup from both measured and model-calculated data.[63] For a discharge burnup in the range of 60-65 GWd/MTU, the maximum oxide thickness is 100 µm and the average hydrogen concentration is 800 ppm, which corresponds to a metal loss of 70 µm using conservative assumptions.[63]

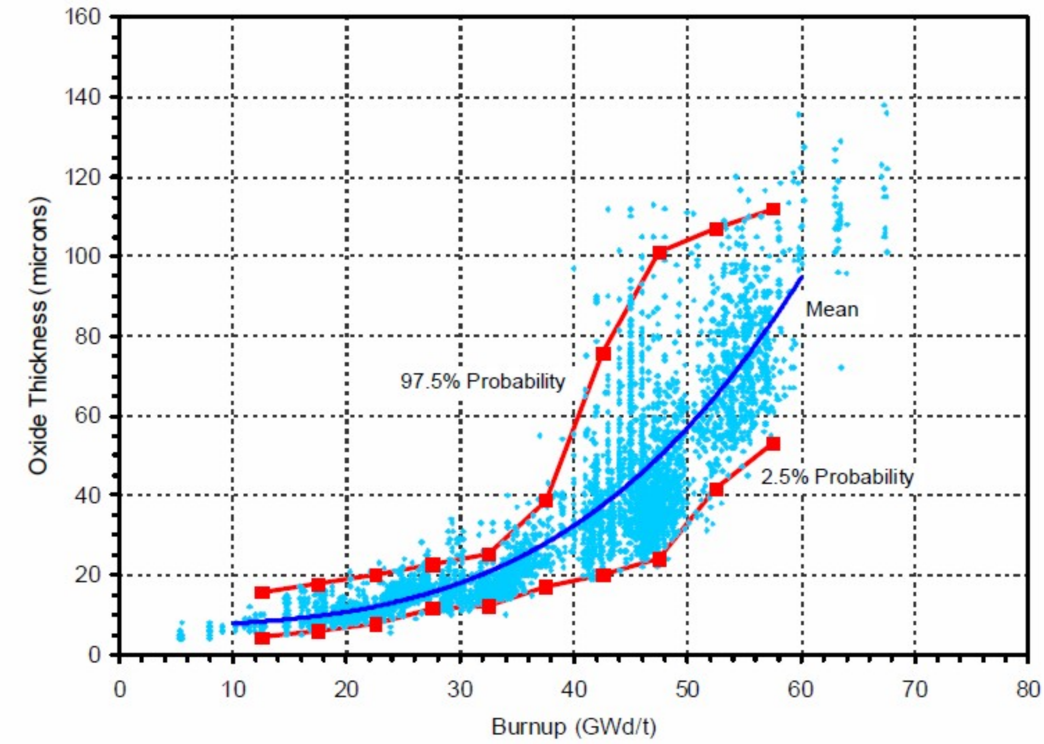


Figure 20. Cladding Outer Surface Oxide Layer Thickness Versus Rod Average Burnup[63]

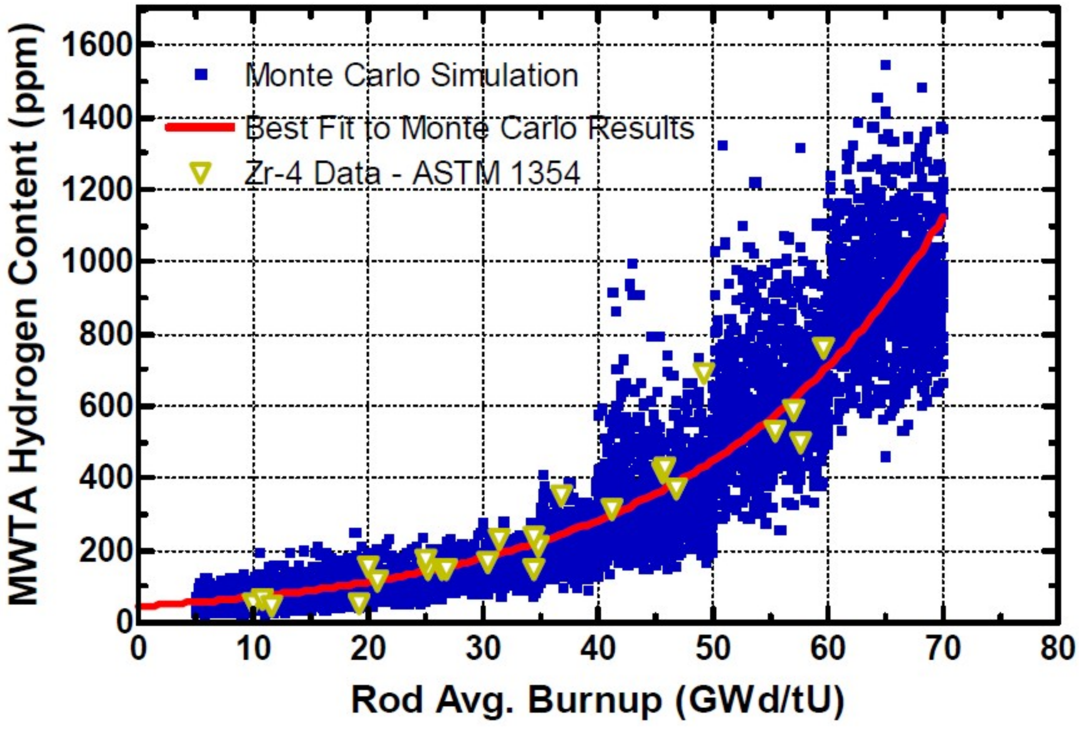


Figure 21. Maximum Wall Thickness Average (MWTA) Hydrogen Content in Low-Tin Zircaloy-4 Cladding[63]

Irradiation in the reactor changes some of the mechanical properties of Zircaloy. Irradiation damage is a complex process involving a variety of environmental factors including irradiation flux, neutron fluence, temperature, and metallurgical variables. Free neutrons that occur as part of the nuclear chain reaction produce point defects and crystal dislocations, which leads to embrittlement of Zircaloy that is equivalent to cold working.[68] In particular, there is an increase of yield strength (called hardening) with fuel burnup, since burnup is directly related to the total neutron flux.[68] The other principal sources of radiation are α-decay of the actinide elements and β-decay of the fission products. Alpha-decay produces energetic α-particles (4.5 to 5.5 MeV), energetic recoil-nuclei (70 to 100 keV) and some γ-rays, whereas β-decay produces energetic β-particles, very-low-energy recoil-nuclei, and γ-rays.[69] In a reactor, α-decay dominates and is the primary source of atomic displacements by ballistic processes in zirconium ceramics. The α-particle and α-recoil-particle are released in opposite directions and produce distinct damage regions separated typically by several microns. Consequently, neutron radiation and α-decay dominate as mechanisms of irradiation damage to cause hardening and embrittlement of Zircaloy while in the reactor.

Some direct physical (wear) damage may occur to the cladding while in the reactor. For instance, the structural grid spacers (zirconium) and supports can cause fretting-wear scars on the exterior surface of cladding during operation[70] (see Figure 22), or foreign material in the reactor coolant such as a piece of wire can be trapped and vibrate against the fuel-rod surface, causing some abrasion of the surface. Fretting wear is the accumulation of friction wear between cladding and other fuel assembly components caused by flow-induced reactor system vibration and load variations under normal service conditions. If wear is severe, a breach can occur; this is the principal cause of cladding failure of LWR fuel rods.[71]

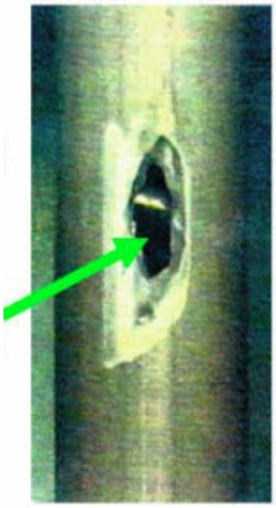


Figure 22. Fretting Wear Leading to Cladding Breach at Grid Spacer[62]

The initial physical condition of the fuel rods and assemblies at the time that they are placed into dry storage is closely linked to their condition when taken out of reactor because degradation during wet storage is minimal. Since it is not desirable to carry out destructive investigation of numerous fuel rods, nor is it practical to inspect all fuel rods and assemblies, even visually, their described condition is based on statistical measures and limited testing. These statistics do not take account of incipient cracks, partial fretting, and other such degradation. Consequently, the typical condition of used fuel is based on operating reactor-records of cladding breaches and fuel that has deformed beyond acceptable limits, ultrasonic inspection (usually only of suspected breached rods), and actual detailed-characterizations of rods performed in hot cells that are destructive in nature. The uncertainty associated with estimating the physical condition of fuel depends on the numbers of actual rods and assemblies evaluated in detail and how representative these rods and assemblies are compared to the entire population of used fuel. In these descriptive statistics, it is probably important to separate used fuel into categories of reactor type (PWR or BWR), cladding type, and burnup (low or high).

Visual examination and characterization of stored U.S. commercial used-fuel was carried out in 1999 and early 2000 at the Idaho National Engineering and Environmental Laboratory (INEEL). A CASTOR V/21 PWR canister type dry-storage cask system was selected to be opened. The used fuel had been stored for about 15 years. The intention of these activities was to inspect and test internal and external surfaces, cask components, a selected number of fuel rods removed from one assembly, and the concrete storage pad. Cask interior crud samples and helium cover gas samples were also collected and analyzed. Personnel at Argonne National Laboratory performed characterization of the fuel rod. Results of the surveillance indicated that the reinforced-concrete storage-pad, the CASTOR V/21 cask, the cask components, and the used fuel were structurally sound and suffered no functional degradation of intended performance.[72] Since this examination of fuel, the results have been used to help characterize the condition of stored used-fuel and cladding for storage and transportation certifications.

Fuel cladding failures of one to several pins have occurred in operating nuclear reactors, but they are not common. Historical reports of in-reactor rod failures indicate that over time failures have varied according to various failure mechanisms. What is not consistently reported is what constitutes a "failure" which makes comparison of data sets difficult. However, as design improvements were made in one area a new mechanism would come to statistically dominate rod failures, but over-all the gross number of failures was never large and the rod failure rates are much smaller today.[73,74] Stainless-steel cladding failures were as high as 1.4 percent, but typically the rod failure-rate was estimated to be closer to 0.2 percent. The failure mechanisms were determined to be fuel-pellet/cladding interaction, stress corrosion cracking (SCC) and debris induced fretting.[52]

In early investigations of Zircaloy cladding, SCC was the most common failure mechanism and caused small hairline-cracks that released fission gases, but not radioactive fuel-pellet material. During the period from 1968 to 1973, Zircaloy-cladding failures were reported that averaged as high as 1 in 100 fuel rods but engineering modifications improved the failure rate more than 10 fold.[75] A 1987 DOE study summarized fuel-rod failure level estimates and indicated that through 1985, there were 1.6 defective rods per BWR assembly, with 90 percent being leakers, for an overall rod failure level of 0.36 percent (1 in 278). The same DOE report said there were 2.1 defective rods per PWR assembly, with 54 percent being leakers, for an overall rod failure level of 0.04 percent (1 in 2,500).[74]

The IAEA stated in 1982 that a rod failure level of 0.01 to 0.02 percent (1 in 500 to 1 in 1000) was typical for PWRs, while for BWRs, a failure level of 0.02 to 0.07 percent (1 in 143 to 1 in 500) "can be maintained."[74] These numbers refer to a fuel-rod breach and not a mechanical operational failure such as rods that bow or twist beyond permitted limits.

A 1997 EPRI study reported that from 1970 through 1985 the historical average of failure level for fuel *assemblies* was 10 to 20 percent and between 1985 and 1996, 0.5 to 3.0 percent.[74] It is not clear as to the definition of a failed assembly, but assuming an average of 144 rods per assembly (55 rods per BWR assembly and 231 rods per PWR assembly) and failure with one rod-failure per assembly, this equates to a 1 in 720 to 1 in 1,440 rod failure-rate from 1970 to 1985, and a 1 in 5,000 to 1 in 30,000 rod failure-rate between 1985 and 1996. A different EPRI study reported the rod failure experience for the year 1989 was 0.004 percent for PWRs (1 in 25,000) and 0.003 percent for BWRs (1 in 33,000). In 2009, EPRI reported at the industry Top Fuel meeting that 87 percent of the U.S. plants were currently operating without any fuel failures. According to EPRI,[76] the final 2008 numbers showed that U.S. plants reported 53 PWR assemblies with failures and 5 BWR assemblies with failures. The approximate fuel-rod failure rates from these assembly failure rates, assuming only one rod-failure per failed assembly, is about 0.002 percent for PWRs (1 in 40,000) and 0.0005 percent for BWRs (1 in 200,000). Generally, failed fuel assemblies are not placed in dry storage, but are retained in wet storage, for as long as the reactor pools are available (until plant retirement). When commercial plants are decommissioned, fuel in wet storage must be transported offsite to another wet pool or placed into dry storage.

While this data is over 10 years old and fuel performance has improved with time, the realities of this inventory of the condition of used fuel needs to be considered for long-term storage.

Summary – cladding state upon entering wet storage. Any one of the phenomena described above and occurring during the 3 or 4 years while in-reactor service can result in damage to or even failure of the cladding tube, but a breach failure is rare. The surveyed types of damage that can occur to fuel cladding include excessive internal gas pressure inducing cladding stresses and strains, pellet-cladding interaction (swelling of fuel pellets that induce stress on the cladding), cladding corrosion, hydrogen embrittlement, stress-corrosion cracking, and fretting wear. During early reactor experience, the commercial fuel-rod in-reactor failure rate was highest and approached 0.4 percent for BWRs, but this rate decreased significantly over just a few years as new cladding types and other design changes were introduced. An assessment of cladding integrity and whether it may be become damaged during the time in dry storage requires knowledge of the initial state of the fuel rods and assemblies when they enter dry storage. The next subsection continues this discussion by focusing on what can be expected to happen to the cladding when it is taken out of the reactor and placed in the reactor storage pool.

4.1.2.2. Wet Storage

As previously discussed, cladding can have manufacturing and handling defects and may become damaged while in the reactor, and so may already have defects and mini-cracks when it comes out of the reactor and is placed in the storage pool. The used-fuel assemblies are lifted from the reactor with a refueling machine and then moved to the storage pool through a transfer tube, all the while remaining submerged in water. Once in the pool, the fuel is placed into storage racks. The fuel assemblies are continuously maintained under water to prevent overheating and to protect workers from radiation exposure. During handling or transportation of used fuel, radioactive corrosion debris, sometimes attached to the outer surfaces of fuel assemblies, may break loose. As the fuel rods cool over time in the pool, the dissolved hydrogen in the cladding precipitates out as hydrides. The internal rod pressure at room temperature is typically 3–7 MPa (440-1,000 psi).[77] During either transfer of used fuel into wet storage or preparing assemblies for dry storage, PWR assemblies are generally inspected visually to identify fuel assemblies that may need to be classified as damaged (and require special handling). The channels (to direct water flow) surrounding a BWR assembly prevents visual inspection, except in cases where they are removed. Video-camera inspection is limited to what can be easily accessed and seen, primarily the peripheral fuel rods and external assembly structures. Any small or "buried" defects are difficult to identify using the video-camera method.

During wet storage, the pool water temperature is maintained at about 40ºC. At this temperature, and over a period of 1 to 3 years, fission gas release will be minimal, there will be no significant thermal pellet-cladding interaction, the cladding will not experience any significant corrosion, and hydride precipitates are stable.[78] No significant further damage to the fuel pellets or cladding is generally expected to occur for intact fuel-rods during their years in wet storage. For the case of rods entering wet storage with existing pin holes or hairline cracks which may contain residual water, radiolysis of this water may occur, producing hydrogen and oxygen, and potentially increasing UO_2 or internal cladding-corrosion.

Researchers have reported minimal to no development of cladding failures up to 20 years of wet pool storage.[79] One set of tests carried out on ten defective fuel-rods with an average burnup of about 30 GWd/MTU and stored in a pool for 6 years did not result in significant further damage (as determined by profilometry, oxide thickness, eddy current testing and visual inspection).[78] On the other hand, "several sites have documented degradation/corrosion of the used-fuel assemblies in wet storage resulting mainly from poor water chemistry control."[80]

Summary – cladding state at the end of wet storage. Cladding may already have some small defects like tiny holes or hairline cracks, internal and external corrosion that has decreased the original metal wall-thickness, absorbed hydrogen, and hydride precipitation; however, it is very rare that new defects are detected while in the pool. Significant cladding defects can be detected during wet storage by monitoring stack off-gas for fission product gas leaks; if leaks are found, then assemblies are further inspected and breached fuel-rods are canned if necessary. Generally, a visual inspection is made of assemblies to identify fuel assemblies that may need to be classified as damaged and require special handling. If the cladding is functionally undamaged, there is an insignificant risk of expected fuel oxidation. Given undamaged cladding and the visible transfer of assemblies into and out of wet storage, the fuel-assembly containment criterion is deemed satisfied. Thus, during wet pool storage, used fuel is not expected to experience significant deterioration before dry storage. *If pool storage of fuel is continued for an extended period, it will be necessary to assess and evaluate the effects on intact or damaged fuel.*

4.1.2.3. The Drying Process

Cask loading operations for CSNF include drying, backfilling with inert gas, and transfer of the cask to the dry-storage pad located outside the reactor building. During this short time the used fuel is subjected to several thermal cycles and mechanical handling forces. The canister or cask must be drained, rendered dry, and backfilled with a cover gas (e.g., helium) before welding (canister) or bolted closure (cask) for subsequent dry storage and/or transportation. When used fuel is removed from the storage pool it will increase in temperature without water to help cool it. If drying is not carried out, water within the sealed storage container could lead to a pressurization problem (steam), a possibility of container or cladding corrosion, fuel oxidation, and other issues.

Details of the drying techniques and process are described in the American Society for Testing and Materials (ASTM) C 1553.[81] The typical drying process consists of draining water from the canister or cask and then removing more water from the interior by applying a repetition of vacuum/backfill drying cycles using inert gas. The degree of drying can be controlled by vacuum level, time, temperatures, and number of drying cycles employed. In addition to vacuum drying, a forced helium dehydration system is used to remove liquid water in the canister, in which dry gas is circulated to evaporate and remove moisture. The forced helium method requires a pressure relief valve to limit the canister pressure and the temperature is controlled to avoid water boiling inside the canister.[81] The drying process employed may not remove all free (unbound) water, especially water held in pores and cracks, within capillary spaces of crud and sludge, and in thin wetted surface films. It is even more difficult to remove water physically or chemically bound to surfaces. Water not removed during the drying process from cladding, container components, crud or sludge, or fuel pellets (if a breach is present in cladding) is named residual water. Residual water has been known to react with fission gas products to form corrosive environments (e.g., caustic hydroxides).[81]

During the vacuum part of drying cycle process, the very low conductivity of the vacuum means the used-fuel assemblies will rise in temperature. The hot fuel helps eliminate residual water. However, the thermal cycles and handling forces have the potential to degrade the used-fuel cladding sufficiently to hinder its ability to meet requirements to be retrievable and remain intact during handling and transport. Consequently, NRC has established operational standards (short-term cladding temperature limits) to minimize this damage.

Temperatures during drying cycles. The peak cladding temperature and hoop stress (due to high temperatures increasing the internal helium gas pressure) during vacuum drying depend on the decay heat load of the CSNF assemblies, the heat transfer characteristics (conduction, convection, and radiation), and the backfill environment of the loaded cask, as does the cooling rate of the CSNF assemblies from the peak cladding temperature. The duration of each vacuum cycle is limited because helium backfill gas is required for providing sufficient heat conduction to cool down the fuel rods as they heat up during a drying cycle. While higher temperatures help in the drying process, temperature limits are set for the cladding and components, which should not be exceeded to minimize the potential of a subsequent radial hydride reorientation and cladding creep.

Figure 23 shows a calculated correlation between PWR-assembly decay-heat-load, the canister surface temperatures and peak cladding temperature (PCT) for a multi-purpose canister that is similar in design to a DOE Transportation, Aging and Disposal (TAD) canister system. The PCTs are calculated as a function of decay heat of the SNF assemblies, the assembly wall temperature, and the interior environment of the cask (i.e., vacuum, helium, or nitrogen (taken as air)). This figure shows calculations for the case of fuel being dried in a vacuum environment which gives significantly higher PCTs than the same calculation done in a helium environment. In a helium environment, and high fuel assembly power (>1,500 W), the calculated PCTs exceeded 350°C in a few cases only. In addition, the modeling showed that the most critical period that could affect the possibility cladding radial-hydride formation is during vacuum drying. The current industry trend of higher-power operation, higher burnup, and shorter cooling time will tend to increase the decay-heat-load in the used-fuel assembly, which would tend to increase the PCT.[83]

Used-fuel cladding experiences a harsher environment for cladding creep in dry storage than it does during reactor operation as a result of higher temperatures experienced by the fuel and cladding, as well as differential pressures across the cladding during the vacuum drying process.[82] Accordingly, the highest historic cladding temperatures are likely to arise during preparation for dry storage or in the early years of dry storage. High cladding temperatures may induce cladding stresses and strain approaching cladding material failure-limits, if they are not properly monitored. The induced high internal-cladding-pressure and high temperatures will likely result in creep. The NRC has set temperature guidance limits for CSNF cladding to ensure cladding integrity. Limits from 400ºC to 570ºC have been suggested (see Section 2).[29]

Potential hydrogen damage. As mentioned above, hydrides precipitate generally in a circumferential direction upon cool down (see Figure 19), which may not be significantly detrimental to cladding mechanical properties. As the temperature increases to the peak value of about 400ºC during drying, the hydrides dissolve into the cladding as hydrogen and the relative internal pressure increases to 6–16 MPa (900-2300 psi).[77] As the fuel then cools, hydrogen re-precipitates out, partially as radial hydrides. The NRC has set peak temperature and thermal cycling guidance limits to minimize precipitation of radial hydrides in cladding during the drying process. Radial-oriented hydrides can embrittle the cladding and lead to delayed hydride cracking. The processes of radial hydride precipitation, embrittlement and delayed hydride cracking are discussed later in subsection 4.4.2.c., “Hydrogen-Related Mechanisms and Effects.” New cladding materials may have very different hydrogen uptake and precipitation behaviors and require continued study, especially their mechanical behavior over the period of extended storage.

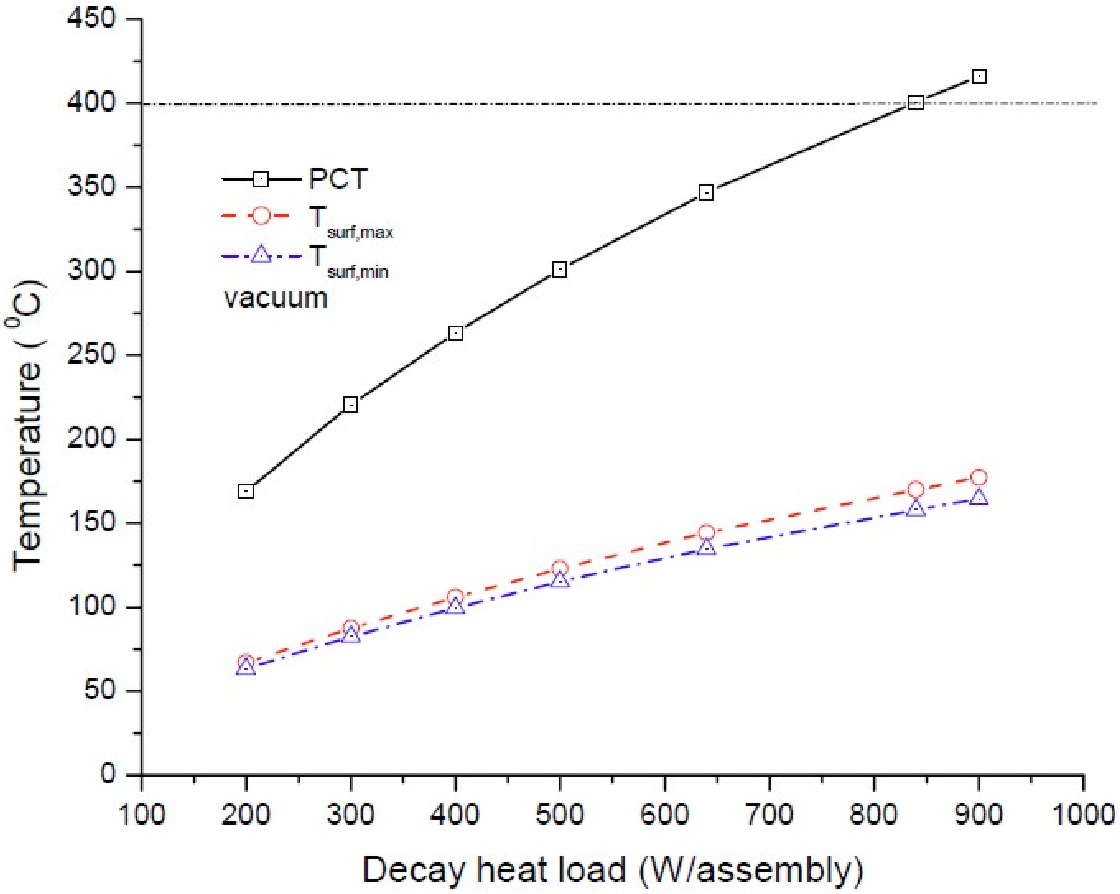


Figure 23. Calculated Peak Cladding Temperature (PCT) and Canister Surface Temperatures as a Function of Assembly Decay Head Load for an Evacuated Canister Containing 21 PWR (15 x 15) Assemblies[83]

Potential fuel oxidation. Before the used fuel is inerted, circumstances may occur during drying and loading, which would allow fuel oxidation to occur. This would be a condition where exposed fuel in "cans" or fuel-pellet material adjacent to minor cladding defects is exposed to air at high temperatures for a sufficient time to oxidize the fuel. NRC ISG-22 (2006), "Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short-Term Cask Loading Operations in LWR or other Uranium Oxide Based Fuels,"[37] gives guidance to help avoid this phenomenon. In part, if there is a possibility of an oxidizing environment, records indicating the time-at-temperature profile need to be kept for fuel rods while they are exposed to an oxidizing atmosphere. With this information an analysis can be carried out to estimate the time required to initiate the rod splitting process. Most oxidation and cladding-splitting studies produced data based on low-burnup fuel and Zircaloy cladding; studies focused on high-burnup fuels and the behavior of new cladding types are needed.

Possibility of cladding breaches related to drying. It is possible that some initially intact fuel-rods may experience further damage under stress conditions caused by handling and transport that may bring them to a minor failure state (e.g., hairline crack in welded zone or cladding body).

The number of minor defects created during drying (if any), how many fuel rods may be losing helium and at what rate, the degree of embrittlement or other degradation characteristics, and change in conditions that may precipitate subsequent dry-storage cladding and fuel-pellet degradation could be better understood and characterized. On the other hand, current industry experience with low-burnup fuel and adhering to NRC restrictions is that very few fuel rods fail during drying, handling, or transport. There is less experience and modeling work to robustly predict what will happen with high-burnup fuels and new cladding types in the conditions of extended storage.

Risk of residual water. Because of the practical temperature and vacuum requirements, some moisture or residual water will remain in the container "as a combination of vapor, free or unbound liquid water, physisorbed water, chemisorbed water, and ice."[81] Water can serve as a moderator, could serve as a source of oxygen for fuel oxidation or container corrosion, or could affect other degradation mechanisms during dry storage. This residual water is categorized into several types. *Unbound water* is water in the solid, liquid, or vapor state, that is not physically or chemically bound to another species. *Chemisorbed water* is water that is bound to other species by forces whose energy levels approximate those of a chemical bond. *Physisorbed water (adsorbed water)* is water that is physically bound (as an adsorbate, by weak forces) to internal or external surfaces of solid material. Ice can form during the drying process. *Bound water* includes adsorbed surface layers of water, and nearly all chemisorbed water. The presence of residual water within a sealed dry-storage cask "may result in container pressurization, fuel retrievability issues, and container corrosion."[81]

Risk of stress corrosion cracking. The incipient cracks on the inner surface of fuel cladding that were mentioned earlier can result in stress-corrosion cracking due to the chemical reaction of fission products such as iodine and cadmium on the cracks. At the tip of the incipient crack in the cladding, bare metal is converted to metal iodide (ZrI_4), which is brittle and facilitates progression of the crack under the existing residual stress regime.[51] Figure 24 shows a typical stress-corrosion crack in the cladding facilitated by a thermally-cracked pellet providing a pathway for fission gases and a non-circular pellet shape because of chipping during fabrication. The stresses that help initiate corrosion cracking arise from the phenomenon of pellet-cladding interaction, which was discussed earlier.

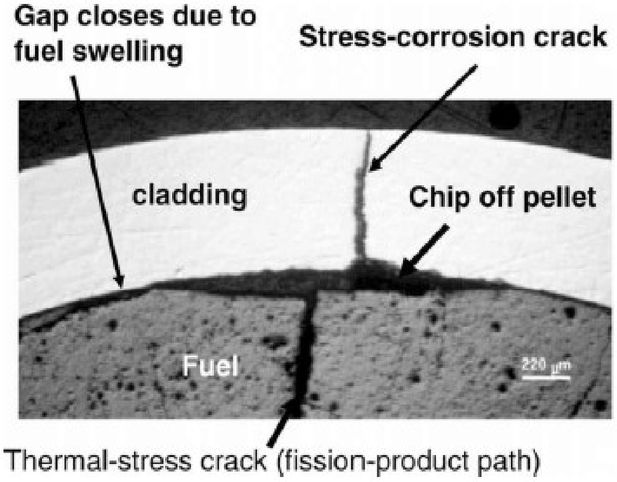


Figure 24. Stress-Corrosion Cracking of Cladding[46]

Summary – cladding state at the end of the drying process: Under current NRC requirements, temperature limits are employed during the drying process, but some degree of damage or change in fuel rod conditions may occur during the vacuum drying process. Furthermore, the statistical characterization of the physical state of the fuel after drying could be improved. This information along with other contextual information (e.g., average burnup), would be helpful in understanding the current level of damage, how damage evolves, and in modeling future condition of the fuel under various conditions. Modeling of the fuel rods and assemblies during the changing conditions of the drying process is carried out to ensure that no significant damage of cladding occurs that may jeopardize its function as the primary barrier to fission product release and retention of helium gas. A more rigorous and systematic inspection or characterization of fuel assemblies before placement into dry storage could be made to help confirm the initial assumptions used in current modeling of drying-process cladding behavior.

Even though no inspection of individual fuel rods is performed after drying, the cladding at the end of the dry-storage process can be assumed to mostly be intact (the cladding is not breached and the fuel rod maintains containment of fission gases and inert helium) . This follows from the observed state of fuel assemblies and rods during wet storage where release of fission gases can be detected. Accordingly, given the NRC temperature limits imposed, cladding can be reasonably deemed to be undamaged (maintains its design functionality, although occasional minor breach defects may exist), which satisfies the fuel-rod confinement criterion. *However, as pointed out previously, cladding may have suffered degradation in the form of incipient cracks and localized thinning that could affect cladding integrity during normal and accident storage and transport conditions.* Because fuel pellets are assumed to be confined in helium (the fuel rods themselves or in canister helium), there is an insignificant risk of fuel oxidation. What is unknown is the degree of residual water remaining in dried fuel, and its ultimate fate and effects. Radiolysis of the water could allow fuel pellet, cladding, or metal component degradation if the water or oxygen/hydrogen gas were to escape cladding through pin holes or cracks. These assumptions and the drying process modeling could be better verified by examination of observations derived from an on-going inspection program that involves a physical characterization of representative fuel rods.

4.1.3. Dry-Storage Systems

This subsection describes parts of the dry-storage system other than used fuel. The properties of these dry-storage system components may also be affected by normal and off-normal conditions encountered during dry storage. Each of these components should be evaluated against changing conditions, degradation mechanisms, and the performance of their intended function(s) over time.

Dry-storage systems are defined as the canister or bare metal cask, the concrete overpack and the concrete foundation storage pad. It is useful to think about the various components in terms of their function in the dry-storage system: (1) used-fuel confinement, (2) criticality control, (3) radiological shielding, (4) heat transfer, (5) structural integrity of cladding and canister including pressurized helium, (6) foundation support, or (7) operations.

4.1.3.1. Metal Canister and Structural Components

As described previously, the two basic types of dry-storage systems are a non-canisterized metal cask system (thick-walled, bare-fuel) with bolted lids and o-rings and a canisterized system in which a welded metal canister (thin-walled) is inserted into a heavy metal or concrete overpack. This subsection refers to all the components that may be present in either type of dry-storage system. Several key components important in terms of long-term degradation behavior are mentioned below.[48]

The metal sealing systems consist of a metal shell with one to three metal lids sealed by multi-pass welds or by bolts with metal seals. The major metal and non-metal confinement components consist of the vessel shell (nodular cast iron, carbon steel, low-allow steel, forged steel, and stainless steel), the lids (304L SS, 193 Gr B7, 320 Gr L43, SB637 Gr NO7718, SA564, Type 630 H1150), and seals. Seals are metal (Inconel X730 spring, Al jacket, Nimonic 90 spring, Inconel X750, Aluminum, 304L SS) and elastomer (ethylene propylene copolymer, silicone rubber, propylene, Viton, and polytetrafluoroethylene or PTFE). Some bolted dry-storage systems also employ neutron shielding materials.

As an example, the Holtec HI-STORM 100 MPC canister system will be described (see Figure 25). The MPC is made entirely of austenitic stainless steel. The 1.27-cm (0.5 in) thick cylindrical shell is constructed with one circumferential and four axial seam welds. The cylindrical shell is then welded to a 6.35-cm (2.5 in) thick baseplate. These are full-penetration submerged arc welds, which undergo dye penetrant and radiographic examinations during manufacture. They are not post-weld heat treated to remove residual stresses. After the MPC is loaded with used-fuel assemblies, the MPC lid is welded to the shell. This 1.9-cm (0.75 in) thick weld undergoes dye penetrant examinations after the root, intermediate, and final weld passes. The vent and drain ports used for hydrostatic testing, drying, and inerting are welded closed with cover plates. These welds undergo a final penetrant examination only. The MPC lid-to-shell weld and the vent and drain cover plate welds form the primary seal of the MPC lid. A closure ring is then placed on the MPC lid and welded to the shell and lid to form a redundant seal. This design ensures that the MPC environment is separated from the outside atmosphere by two independent lid welds. It should be noted that the procedures described above do not represent regulatory guidance nor are necessarily typical of industry practice.

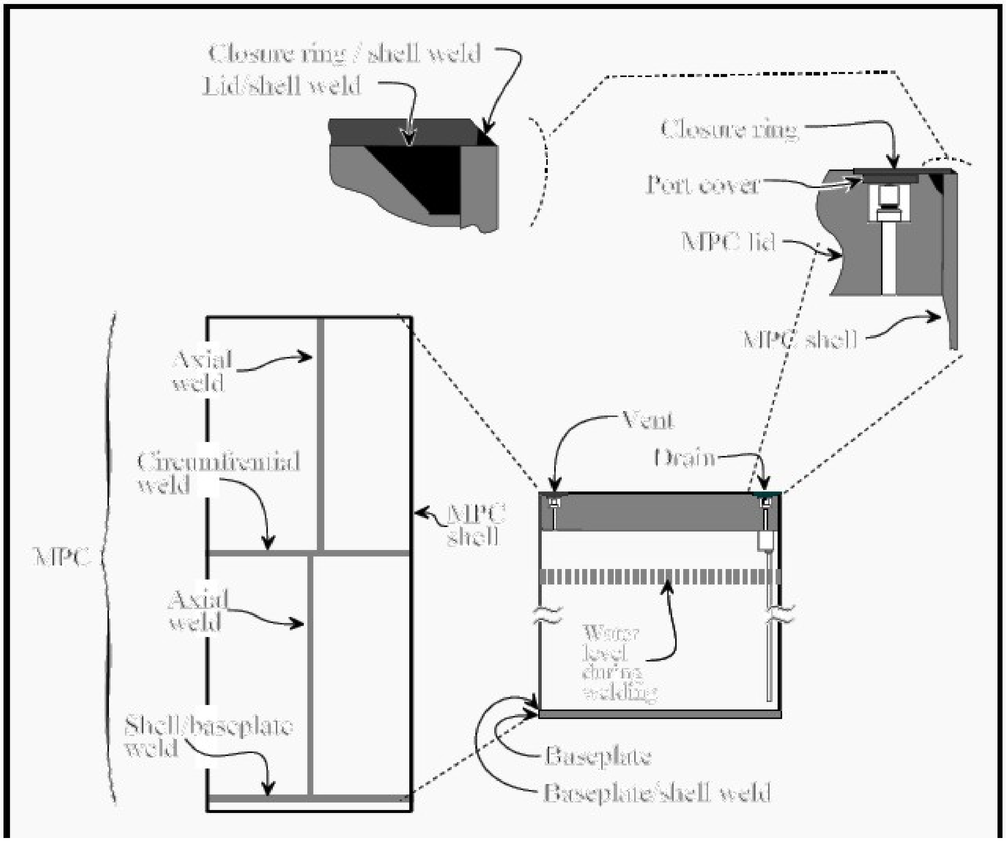


Figure 25. Drawing of Holtec HI-STORM 100 MPC Construction with Weld Locations (best available figure) [84]

The basket assemblies (basket and neutron absorbers) are the main components functioning as criticality control. The basket assembly itself is made from 304 SS with boron, 6061-T651 Al, SA 705 Type 630 SS, Inconel, SA 516 Gr 70, 240SS, and 479 SS. The neutron absorbers are made from metal matrix composites such as boron carbide/Al matrix with Al cladding and castable borated shielding. In recent years, there has been a trend toward using more corrosion resistant borated stainless steel and other metals for neutron absorbing. The metal rigid structural skeleton of the fuel assemblies helps prevent the fuel rods from bending and buckling in a handling or transportation accident.

The shielding function consists of gamma and neutron shielding. The gamma shielding materials include thick-walled steel and iron shells, ASTM B29 chemical lead placed between the inner and outer shells, and reinforced concrete. The neutron shielding materials consist of polyethylene, polypropylene, and borated polymer shielding.

In the example of the Holtec HI-STORM 100 cask system, the fuel basket is free standing, but held in position inside the MPC by basket supports. Boral plates along the walls of each cell prevent the used fuel from becoming critical should the canister fill with water as a result of a severe accident, while upper and lower fuel spacers keep the fuel assemblies vertically positioned in the basket. Except for the boral plates and heat conduction elements, all MPC components are made of stainless steel.[84] For operations, lifting lugs and trunnions or grapples are made mostly from stainless steel metals. Several penetrations may extend through the confinement shell or lids including the drain port plug, vent port plug, pressure port plug, interlid port plug, and pressure sensor. Most of these are made from stainless steel and may include PTFE O-rings. Pressure sensors are typically electronic with a Ni-plated Cu-Be diaphragm.

The above and similar internal materials are subject to degradation over time depending on the internal environmental conditions. Consequently, special attention should be placed on dissimilar materials that are in contact and coatings. Examples of dissimilar materials in contact include:

- Metallic or elastomer O-rings (Al, SS, Inconel, ethylene, propylene, silicone rubber, PTFE, Viton) in contact with a sealing surface (Ni or SS shell or lid)
- Shielding material (Boral, Boraflex, Carborundum, BISCO, Metamic, lead) in contact with structural components (assembly grids, shell, basket)
- Coated material (electroplated Ni 1.5 mm thick, Zn/Al, Ti/Al, Everlube 812 & 823, CarboZinc 11, Dimetcote 6, SS weld overlay, epoxy resin, paint, and Al oxide) in contact with non-coated or coated material (inner cask surfaces, O-ring surfaces, fins, and basket assembly).

4.1.3.2. Inert Fill Gas

Importance of Inert Fill Gas (Helium). Because helium is not corrosive and has a high thermal conductivity, it is used as an inert fill gas in both fuel rods (within the cladding) and fuel storage canisters or containers. Under a normal atmosphere containing oxygen, minor breaches (pinholes and hairline cracks) in the cladding could develop into gross ruptures because of fuel oxidation at early dry-storage temperatures. A helium atmosphere during dry storage is used to prevent the possibility of fuel oxidation and gross ruptures due to unzipping, and corrosion of internal canister components. Oxidation of the fuel would also violate regulatory requirements in that the chemical form of the UO_2 fuel-pellets, as listed in the certificate of compliance, would change. The typical helium fill gas pressure of LWR fuel rods is between 1.5 to 3.5 MPa (220 to 510 psi) at 20ºC for PWRs and between 0.3 to 0.7 MPa (44 to 100 psi) at 20ºC for BWRs.[48] A more recent EPRI report showed representative PWR internal rod pressures at the start of dry storage to lie between about 3 to 4.5 MPa (440 to 650 psi) at 25ºC. Oxidation caused by fill gas impurities is assumed to be negligible.

a. **Loss of fuel-rod fill gas before dry storage begins.** The only way that a significant amount of fuel-rod fill gas could be lost would be through penetrations in the cladding. Such pinhole leaks or cracks would allow radioactive materials to escape into the reactor pool, which could be detected by normal monitoring while in the pool. The fuel assembly containing the leaking rod would be identified by sniffing for fission product gases.

b. **Extended dry storage leakage of fuel-rod fill gas.** Loss of fill gas from a single used fuel-rod during storage is unlikely to cause temperature problems or lead to fuel-pellet oxidation as long as the cask fill gas (in the case of a bare-fuel storage system) or canister (in the case of a canister-based storage system) containing the assembly with the failed rod has not escaped. However, operations following dry storage, such as transferring the used fuel into a transportation cask or transferring used fuel to a disposal container, may be problematic due to the release of radioactive fission gas into the canister fill gas.

c. **Effect on fuel-rod temperature of loss of fuel-rod fill gas during dry storage.** If the helium fill gas in the cask or canister is present, the effect of losing some of the higher pressure fill gas from a fuel rod is expected to be small. Heat transfer to the cladding wall in the depressurized rod would be lower than for an intact fuel-rod, but the bulk inert gas of the canister would provide a large thermal sink that would dampen the higher temperatures of a failed fuel-rod.

d. **Loss of fill gas in bare-fuel (bolted) storage systems.** A penetration in the cask container of a bare-fuel storage system would result, over time, in the dilution and eventual replacement of the fill gas in the cask by air. Air can lead to oxidation and corrosion, particularly when the temperatures of the fuel-rod surfaces fall below the dew point of the air and particularly if the storage system is located in marine, industrial, or other environments where corrosion agents are present in the air. In addition, the thermal conductivity of air is approximately one-sixth that of helium, and fuel-rod temperatures would be higher in an air environment compared to a helium environment. Bare-fuel based storage systems are regularly monitored for the presence of helium fill gas, and appropriate actions are taken (replace the fill gas and repair or replace the cask container) when leaks are detected. In October 2010, a TN-68 vertical dry storage cask at Peach Bottom was found to be leaking helium from its main outer lid seal and the fuel assemblies in the cask will have to be unloaded for repairs. The leak was detected by the pressure monitoring system and was determined to be "very small."[85] As long as regular inspections for the continued presence of fill gas are carried out, no obvious containment research is needed unless occasional inspection of representative used fuel dry-storage systems reveals indications of potential cask breach failures. This assumes that a material loss balance analysis has been done (rate of fill-gas replacement by air as a function of penetration defect size and time) to determine an appropriate inspection frequency.

e. **Loss of fill gas in canister-based storage systems.** A penetration in the canister wall of a canister-based storage system would result, over time, in the dilution and eventual replacement of the fill gas in the canister by air. Loss of fill gas depends on the pressure inside the cask, the size of the breach, the temperature within the cask, and the amount of fill gas. For a generic dry storage canister, Casella calculated depressurization times ranging from almost 500 years to 8 months for a cylindrical pinhole breach radius varying from 5 µm to 35 µm, respectively.[86] It is assumed that canisters in canister-based based storage systems are not directly monitored for the presence of helium fill gas over time except at the time that they are filled and sealed (welded). Indirect monitoring in the form of radiation detectors at or near the pad can be carried out, such as a technique developed by the Japanese.[87] If a used fuel-rod has one or more defects, small amounts of fission gases are released by diffusion into the canister, including Kr-85 gas. Research indicates that gamma rays at an energy level of 514eV are characteristically emitted from Kr-85 and should be detectable outside the sealed canister against the high background gamma ray noise.

Optical or nondestructive testing methods may be adaptable to examine external canister surfaces, particularly the welds for defects. In addition, nondestructive tests could possibly be employed to reveal useful information about the degradation of internal components, including the characteristics of internal fill gas. Focused research may discover helpful new nondestructive test methods. If a crack or other defect was to fully penetrate the canister wall, the fill gas could eventually be replaced with air. Fuel rods would subsequently experience higher temperatures than if the fill gas had not escaped and conceivably damage to the rods could result from the higher temperatures. To limit fuel oxidation, assurance must be provided that the helium cover gas remains in the canister as long as fuel temperatures remain high. In addition, the assured presence of helium would prevent the initiation of metal corrosion mechanisms after air enters the canister.

4.1.3.3. Concrete Structures and Components

Reinforced concrete structures and components fill multiple roles as part of a dry-storage system. The major structures are the overpack or radiation shielding enclosures and the concrete foundation pad. For canister-based storage systems, the concrete shell not only provides shielding but ensures the stability of the upright canister, facilitates anchoring, allows ventilation, provides physical protection against attack, theft, and severe weather events. The massive reinforced concrete foundation pad is carefully designed to handle site-specific seismic motions, lateral loading moments, and bearing capacity problems. The structural design of other structures and components is intended to withstand normal operating conditions, off-normal conditions, and accident including extreme natural phenomena conditions such as hurricanes, tornados, and earthquakes.

As an example, the Holtec HI-STORM 100 storage system will be described.[84] The system consists of an MPC that confines the fuel assemblies, a transfer cask that shields workers from radiation while the cask is being prepared for storage, and a storage cask that shields people from radiation and protects the MPC during dry storage.

The HI-STORM 100 transfer overpack consists of an inner steel shell, lead shield, and outer steel shell, and a water jacket provides additional shielding. The pool lid at the bottom of the transfer overpack seals the inner cavity, thereby retaining water to prevent the exterior of the MPC from being contaminated. A lift yoke attaches to lift trunnions on the transfer overpack.[84]

The HI-STORM 100 storage overpack consists of inner and outer steel cylindrical shells connected by four steel, rectangular, full-length, radial plates that extend from top to bottom. The volume between the shells is filled with unreinforced concrete. The shells and the radial plates form the structural member. The concrete is for radiation shielding. The lid is bolted to the overpack at anchor blocks, which are attached to the radial plates. Sixteen open vertical channels are situated around the MPC, between the overpack and the MPC, to allow air to circulate; entering the inlet vent at the bottom of the overpack and exiting the outlet vent at the top of the overpack. The steel channels are designed to progressively collapse during a severe impact, absorbing some of the impact load in the axial direction that could otherwise be transmitted to the MPC. Shield cross-plates next to the vents decrease radiation streaming through the vents. Screen covers prevent insects, birds, and animals from entering the vents to build nests that might block the vents.[84]

Summary – as built cask system: Given that a dry-storage system may need to remain functional for 60 years or longer, over this time horizon; each of the storage system components would be expected to age and deteriorate, and will need to be evaluated with respect to their ability to perform their intended design function(s). The key to successful long term dry storage is to ensure that the helium cover gas remains in the canister and that all degradation mechanisms affecting the ability of the canister to maintain its integrity are minimized. Concrete aging and degradation can be routinely monitored and repaired as needed. Extreme weather or natural phenomena will need to be addressed, recognizing the slow but consistent environmental degradation of the storage system. The canister welds are of particular interest to document and inspect for evolving cracks and other defects, along with monitoring the continuing presence of canister helium.

4.1.4. Current and Future Trends

New materials, techniques, procedures, and regulations. A variety of new fuels, claddings, assembly geometries, and burnups have been, are, and will likely continue to be placed into extended storage. For example, since high-burnup fuels require improved cladding performance, new cladding materials have been developed and employed by vendors such as Areva's M5 alloy, Westinghouse's optimized ZIRLO, Siemen's Duplex, and Mitsubishi's M-MDA material. These new cladding types already form a significant part of the used-fuel population and their use will increase in the future. Most of the research and the design basis to date have focused on in-reactor performance of the cladding: low corrosion properties, robust as to mechanical behavior, effects of irradiation, coolant chemistry, and effects of sudden cooling-water quench. EPRI has supported an important research program for about a decade with the purpose being to provide the technical bases and supporting experimental data needed for dry storage and transport of high-burnup used fuels.[88] However, a better understanding of the specific behavior of these new materials under low temperatures and extended dry-storage conditions is still a research need. As longer-term dry-storage aging and degradation mechanisms are better understood, it is likely that specific conditions found to be important and combinations of mechanisms will lead to new research concerns.

The realities of terrorist threats (see subsection 4.3.3), the need to store CSNF for many more years than anticipated, the desire to implement probabilistic risk assessment methodology and to simplify complex regulations regarding storage and transport of used fuel, and the increasing utility costs to store CSNF for an extended period; provide motivation to consider and implement new types of designs and utility. For example, economic advantage of fuel-cycle costs and reactor operation has and continues to drive the nuclear power industry to burn fuel to higher burnups. There is a clear trend toward phasing out low-burnup fuels and using higher-burnup fuels. Figure 26 and Figure 27 (blue is low-burnup and red is high-burnup fuels) show how this trend has changed over the last decade and will continue in the future.

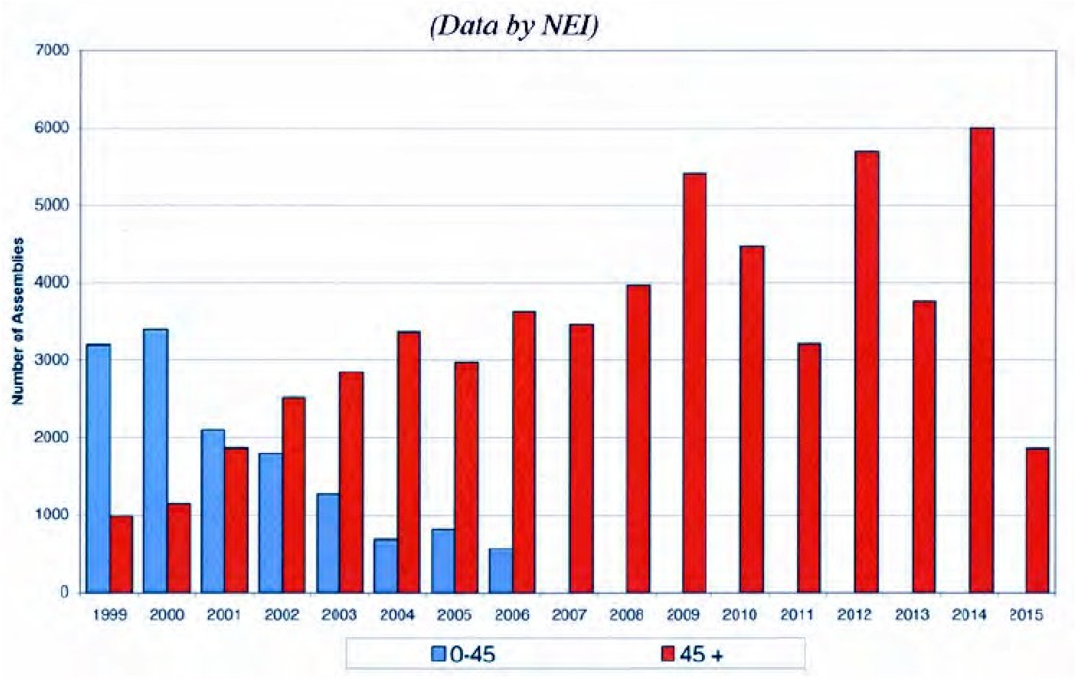


Figure 26. Projected BWR Burnup Distribution Through 2015[89]

(Note: graph units are GWd/MTU)

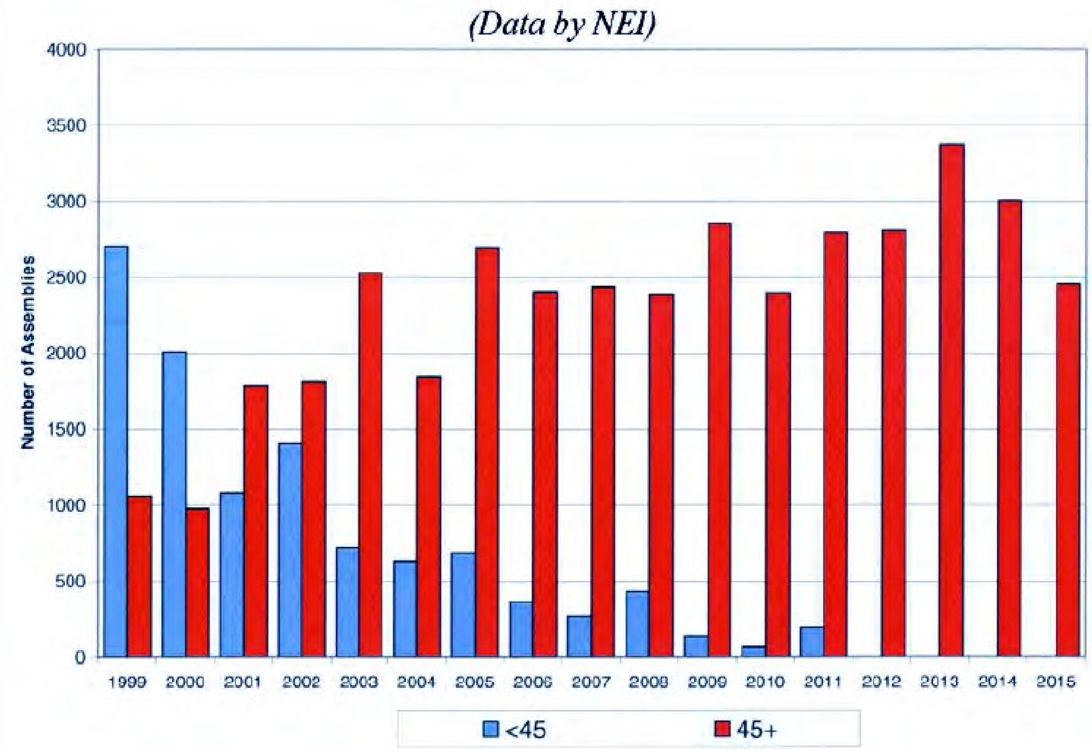


Figure 27. Projected PWR Burnup Distribution Through 2015[89]

(Note: graph units are GWd/MTU)

Effects of switching to high-burnup fuel. High-burnup fuel has a greater initial enrichment, more radioactive decay heat, and higher neutron and gamma sources. The increased irradiation and heat production that come with higher burnups can result in more initial-state fuel-pellet damage, more released fission product gases into the plenum of the fuel rod, and higher fuel and cladding stresses. The higher decay heat loading of used-fuel assemblies will accelerate the rise-of-peak cladding temperature during drying. On the other hand, the outer edges or rim of high-burnup pellet material has smaller grain size, more porosity and higher fracture toughness. This produces two key effects once the rim develops: (1) only a few percent of fission gas are released in the fuel rod void space - most is retained in the porous rim; and (2) the rim becomes resistant to breaking down into fine particles.[42] Increased burnup will increase the corrosion in zirconium alloy cladding because higher burnup generally implies longer in-reactor residence time. PWR cladding is likely to experience even more corrosion due to the higher temperatures that accelerate the corrosion rate (the BWR cladding corrosion rate is much less dependent on temperatures). With increased burnup, more corrosion-produced hydrogen will lead to more hydrogen being absorbed by the cladding. More hydrogen pickup will lead to more hydride precipitation and possibly other effects such as more embrittlement, delayed hydride cracking, and acceleration of the corrosion rate.[90] Many of these factors may significantly affect the physical state of the used fuel and cladding during dry storage. EPRI is studying the maximum amount of hydrogen pickup that occurs for various types of cladding at various burnups and using this to model the amount of hydrides expected. Additional research is needed to better quantify and understand the long- and very long-term effects of switching to high-burnup fuel.

Summary – current and future trends: High-burnup fuel is now queuing up to move into dry storage. There is currently limited data available related to the effects of high-burnup fuel on the physical state of fuel and cladding over time, and effects on other dry-storage components. Dry casks are being loaded with much higher thermal loads and more used-fuel assemblies. These cask systems need to meet the same storage criteria but they have not been analyzed for long-term storage and degradation mechanisms. New promising materials and designs are appearing that can be studied in hopes of improving the extended storage performance of dry-storage systems and aged-fuel safe transport.

4.2. Normal Operations and Transient Conditions During Dry Storage

Normal Events

Storage phase. The vertical or horizontal storage cask systems remain in place, resting by gravity, on the storage pad. Currently, the storage casks do not have (and do not require) instruments or other mechanisms to monitor heat loads or radiation leaks from the MPC. Routine surveillance ensures that the vents remain unblocked and radiation levels are taken during these inspections to confirm existing conditions.

Normal Dry-Storage Conditions and Operations. Specific normal conditions to be addressed at a particular licensed dry-storage site are documented in a Safety Analysis Report for that dry-storage system and refers to credible environmental conditions that fuel may experience during dry storage. Normal conditions include a dry and inert protective environment for: (1) the CSNF and the inner and outer surfaces of the cladding, and (2) inside the cask and all interior components of the storage container. Neutron absorbers must continue to be adequately effective and structural components (e.g., baskets, supports, weld closures, lifting lugs) of the specific storage system must have sufficient strength to meet the required performance. Any seals must be maintained as specified in the Safety Analysis Report. Normal operations comprise movement of fuel to the storage pad and any other handling. These operations may cause slightly bumping and vibration that is felt by the fuel and assemblies.

Radioactive decay and temperature over time. The primary transient condition during dry storage is the radioactive decay of fission products, activation products, actinides, and actinide daughters. As a consequence, both the radioactivity and thermal heat generation decrease over time (see Figure 28, Figure 29, and Figure 30). The radioactivity and heat are dominated by fission products for about the first 100 years, and then the dominant characteristics transition to actinides over the next few centuries. The source of the radioactivity is because of the presence of 3 to 4 atomic percent of fission products (e.g. ^{129}I, ^{131}I, ^{137}Cs, ^{90}Sr), transuranium elements (e.g. ^{239}Pu, ^{237}Np, ^{241}Am), and activation products (e.g. ^{14}C, ^{60}Co, and ^{63}Ni) in the metal used-fuel assemblies. The very penetrating ionizing radiation (β and γ) comes mainly from short-lived fission products, such as ^{137}Cs and ^{90}Sr, with half-lives of about 30 years. These fission products are mainly responsible for the thermal heat from the fuel. The less penetrating radiation from α-decay events comes mainly from the very long-lived actinides, such as ^{239}Pu and ^{237}Np, with half-lives of 24,100 years and 2.1 million years, respectively.[5]

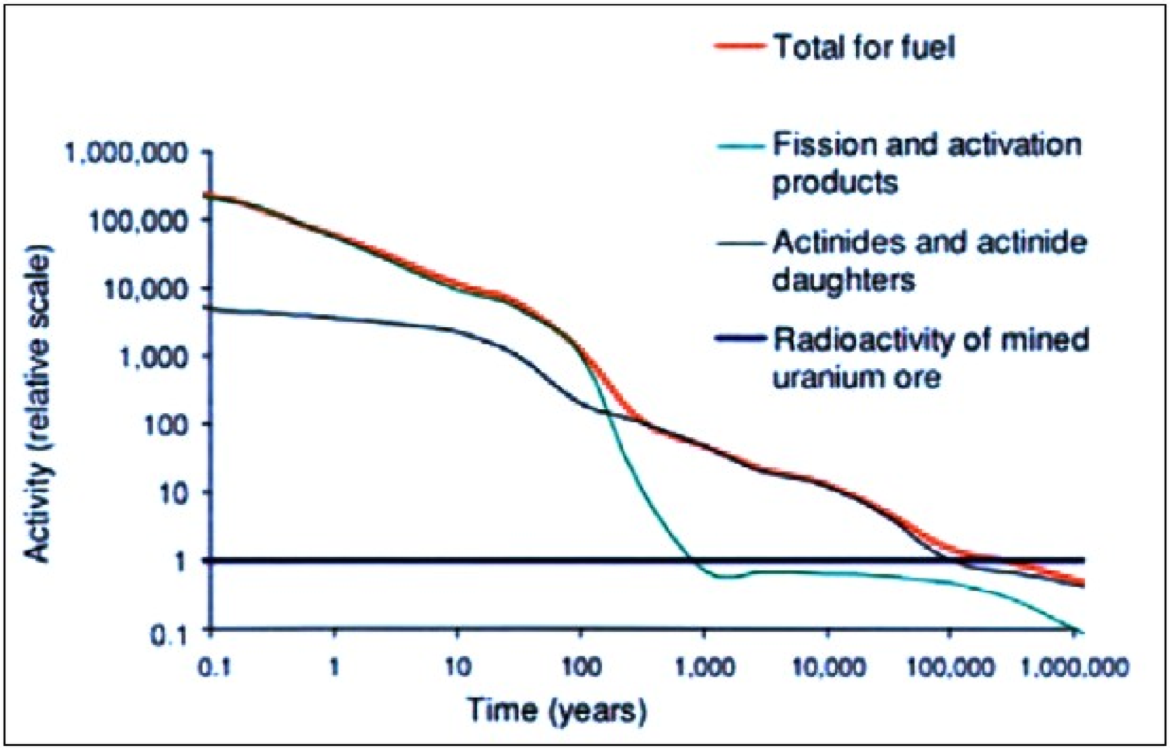


Figure 28. Relative Radioactivity of Used Nuclear Fuel with a Burnup of 38 GWd/MTU[5]

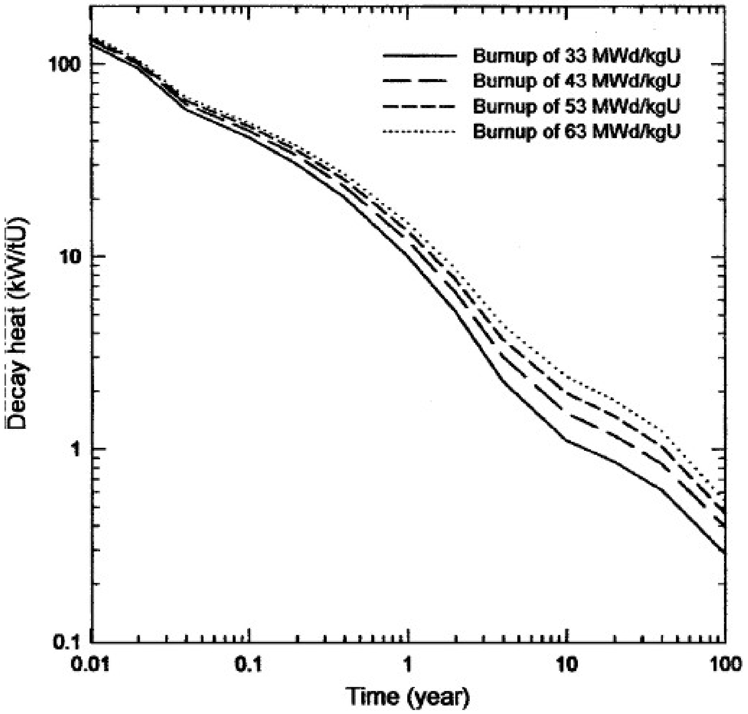


Figure 29. Decay Heat as a Function of Time from 0.01 Years (about 4 Days) to 100 Years for Used-Fuel Burnups of 33, 43, 53, and 63 GWd/MTU Where 33 Was Typical for the 1970s and 50+ for Current Burnups[91]

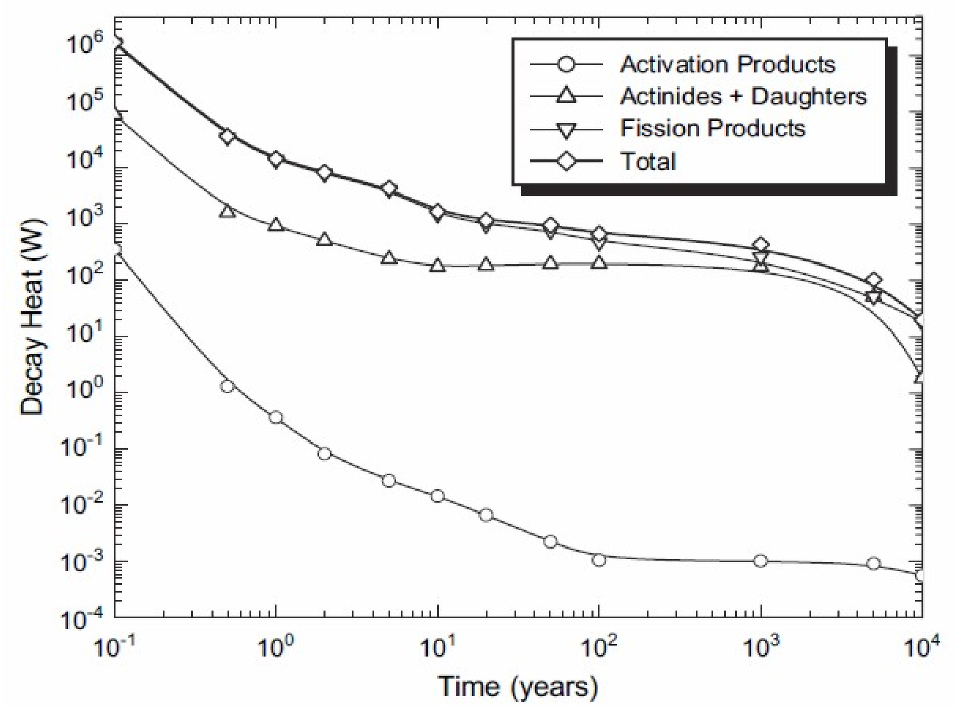


Figure 30. The Variation of Decay Heat for 10,000 Years for a Typical 30 GWd/MTU Burnup PWR CSNF[84]

Temperature is an important factor for material performance because many degradation mechanisms are thermally activated. Over time temperatures will decrease, but the magnitude will depend on fuel burnup, initial enrichment, decay time, and the design of the specific used-fuel storage system. In general, cladding temperatures are expected to drop from about 400ºC to 100ºC over the first 10 years (assuming 30 GWd/MTU burnup, 5 year cooled fuel), with the temperature remaining at about 100ºC for the next 90 years.[48] An EPRI numerical simulation of cladding temperature at the hottest axial position of a fuel rod with an assumed initial peak cladding temperature limit of 400ºC prescribed by ISG-11 Rev. 3 showed that after 10 years of dry storage the rod temperature was about 300ºC, 260ºC after 20 years, 235ºC after 30 years, and 220ºC after 40 years.[63]

The decay heat and radioactivity of the used fuel are due to fission products, actinides and daughters, and activation products. Actual temperatures experienced by the fuel and cladding may vary somewhat because of the effectiveness of the passive cooling of storage systems and their interaction with the storage site weather conditions. Figure 29 shows the decay heat curve for a period of 100 years generated by several different burnups of irradiated fuel. Within a year of discharge from the reactor, short-lived radionuclides decay away, dropping heat generation by a factor of 100 to a value of about 10,000 kilowatts of radioactive hear per metric ton of uranium (kW/tU). After 5 years, the decay heat is down to 2-3 kW/tU, which is about the time the used fuel might enter dry storage. After 20 years in dry storage (25 years total), the decay heat ranges from about 0.7 to 1.5 kW/tU, depending on the original burnup. The decay heat curve over a longer period of time is shown in Figure 30, which also gives component contributions.

Fission products initially dominate the thermal power generated, then actinides reach an equivalent output. The contribution from the activation products is small and barely exceeds 2 percent of the total. During the period (100-1000 years), the actinide isotopes of importance are ^{238}Pu, ^{239}Pu, ^{240}Pu, [and] ^{241}Am…The major contributor to the total radioactivity after one year of storage includes four separate decay chains (^{90}Sr → ^{90}Y, ^{106}Ru → ^{106}Rh, ^{137}Cs → ^{137}Ba, and ^{144}Ce → ^{144}Pr), one additional cesium isotope (^{143}Cs), and one actinide (^{241}Pu). After 100 years, the total activity will have decreased to 132,000 Ci, with the fission products (^{90}Sr → ^{90}Y, ^{137}Cs and ^{137}Ba) supplying ~ 87 percent of the total.... The output of ORIGEN gives the total heat generated by the used-fuel assembly [one-third core or 40.3 assemblies] after 10 years decay to be 20.4 kW and the heat generated by a single fuel rod [204 fuel rods per assembly] to be 2.44 W.[84]

In summary, over the period of dry storage, as the radioactive decay heat decreases with time, the internal fuel-rod gas pressure and cladding temperature generally both decrease. In addition, overpressure still leads to the internal free volume of the fuel rod increasing as the cladding creeps outward (continues until internal pressure is no longer able to induce outward creep), decreasing the gas pressure and cladding stresses further. All of these factors eventually decrease the traditional creep rate to a negligible value; however, there may be exceptions. For example, in the case of the accident scenario of blocked ventilation, temperatures can still spike. Figure 31 shows a centerline temperature profile (numerical simulation) from top to bottom for a Holtec HI-STORM 100 dry-storage system with blocked vents. Also a low temperature, low pressure, creep mechanism may be active over the long-term.

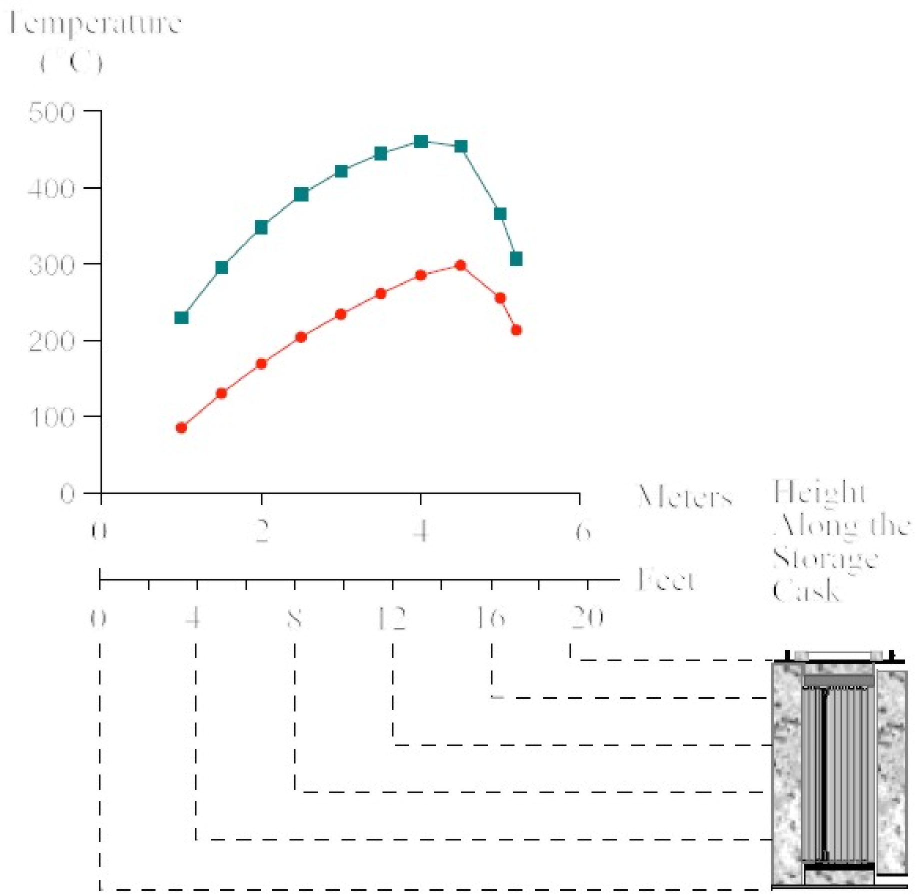


Figure 11. Centerline Fuel Temperature
• Normal Operating Temperature
▪ Temperature with Blocked Vents

Figure 31. MPC Centerline Fuel Temperature as a Function of Height
(unknown fuel age, best available figure)[84]

Canister pressure over time. Pressure inside the canister or storage container may increase or decrease with time. There is an initial backpressure from the helium backfill. The reduction of temperature profile as a function of time will tend to decrease the canister pressure. However, over time the amount of free water in vapor form, gaseous radioactive decay products, and radiogenic gas formation rates can all increase or decrease canister pressure, depending on the circumstances. As a worst case calculation (assuming only reactions that increase pressure), "such calculations yield values of up to 0.35 MPa (50.8 psi) internal pressurization of commercial fuel container from hydrated uranium oxides [assuming 1 kg of $UO_3{\cdot}2H_2O$ which yields 112 g of water] heated to

250ºC by decay heat after sealing."[81] This should not be high enough to cause any significant problems.

Chemical environment over time. The chemical environment that may be active during dry storage includes the backfill gas employed, residual water, zinc vapor if internal components are galvanized, and potentially fission products from cladding breaches. The drying process does not remove all the water in the storage container for dry storage of CSNF. ASTM Standard C 1553 – 08[81] discusses the forms of residual water found in CSNF containers and the potential effects of residual water on CSNF and containers: "After drying, residual water in a variety of forms may remain on the fuel, fuel cladding, or internal components of the container….Water in the container can be released by direct decomposition of the chemically bonded species, vaporization of the physisorbed and the free water, and radiolytic decomposition. These forms of water and decomposition products could cause corrosion, pressurization and possibly embrittlement issues for the storage of spent fuel."[81] The residual water is expected to be quite low if drying was properly performed.

Aspiration of water on metal surfaces. Finally, dry-storage systems employed for CSNF are of the passive vented type, meaning outside air circulates next to the external surface of the metal canister to cool it. For an intact canister, as the canister cools, it may condense water and other particles on its surface from the external atmosphere as a function of external temperature and dew point. *If cracks or holes* develop in the canister, and as the fuel continues to cool, water may ingress to reside on the external surface of the cladding. Aspirated "water ingress during 'dry' storage may significantly increase the overall chemisorbed water content of the SNF over the storage period, especially if the SNF is badly damaged."[81] So there may be a point in time when a significant source of chemisorbed water appears on the surface of the fuel rods. Free water "can be released by direct decomposition of the chemically bonded species, vaporization of the physisorbed and the free water, and radiolytic decomposition. These forms of water and decomposition products could cause corrosion, pressurization and possibly embrittlement issues for the storage of spent fuel."[81] It is expected, however, that if holes or cracks develop in the canister, it will be able to be measured in terms of loss of helium and increased radioactive releases from the cask system and remedial actions such as re-packaging could be taken.

4.3. Significant Storage Events to Consider

A dry cask storage system is designed to withstand man-made events due to handling accidents and external natural events, such as earthquakes, tornadoes and floods, without impairing its capability to perform its intended design functions. The regulations in 10 CFR 72 that require the applicants to design structures, systems and components important to safety to withstand the man-made and natural events[6] were described earlier in Section 2, "Requirements for Dry Storage and Handling." Because a dry-storage system is a free-standing structure, one of the failure modes could be a tip-over event that causes an impact collision with the concrete support pad. The Standard Review Plans NUREG-1536[24] and NUREG-1567[92] provide the NRC staff guidance in reviewing the license applications.

4.3.1. Normal and Off-Normal Conditions

Normal conditions are the primary expected external conditions that affect the dry-storage system performance, including: ambient temperatures, solar radiation energy (insolation), weather, and the general operational environment. The operational environment includes the normal handling conditions such as: the manner in which the cask is loaded, unloaded, and lifted, or the manner in which fuel assemblies are unloaded and reloaded. Normal operations for storage comprise movement of fuel to the storage pad and any other handling. These operations may cause slightly bumping and vibration that is felt by the fuel and assemblies. Normal transport operations may involve much higher levels of vibration and bumping and for longer duration.

Off-normal events and conditions refer to the maximum level of an event or condition that, although not occurring regularly, can be expected to occur with moderate frequency, and for which there is a corresponding maximum specified load, acceleration, or other requirement. Specific off-normal conditions to be addressed at a particular licensed dry-storage site are documented in a Safety Analysis Report for that dry-storage system. NUREG-1536 indicates that off-normal conditions that should be examined "include variations in temperatures beyond normal, failure of 10 percent of the fuel rods combined with off-normal temperatures, failure of one of the confinement boundaries, partial blockage of air vents, human error, out-of tolerance equipment performance, equipment failure, and instrumentation failure or faulty calibration."

Any off-normal events that actually occur during a licensed storage term must be evaluated for their affect on materials behavior and capabilities to meet functional requirements in the future. In particular, five material evaluations need to be performed: (1) factors affecting performance, (2) used fuel, (3) cladding, (4) cask components, and (5) concrete structure, pad, and foundation.

4.3.2. Accident Conditions: Manmade and Natural Events

Accident conditions include both design-basis accidents and the design basis for events and conditions caused by natural phenomena. A hazard analysis study is to be carried out at each site and for each dry-storage system, and all credible accident hazards analyzed. Some accident conditions and events may damage components of the dry-storage system, therefore maintenance or predesigned engineered preventative measures could be employed.

NRC conducted a hazard analysis study related to the handling and transfer of fuel assemblies from the reactor pool into dry storage. They identified 34 different stages and identified the height of the transfer cask, direction moved, rigging of the case, and surface over which the cask was moved in order to carry out the risk-hazard analysis to determine the worst case initiating events to study. In one hypothetical accident scenario they assume a loaded but open canister is dropped (5 ft through the air and 37.5 feet through water) to impact the concrete at the bottom of the storage pool. They determined that "100% of the fuel cladding is breached... and that the probability of a release from the MPC is 1.0." This type of detailed risk analysis is beyond the scope of this study; but it illustrates one approach to assess risks from various scenarios.

NUREG-1536 recommends that the following accident conditions should be considered in evaluating the dry-storage system: cask tip-over, fire, fuel-rod rupture, cask leakage of the confinement boundary, explosive overpressure of canister, and air flow blockage. Other credible accidents that may be derived from a hazard analysis could include accidents resulting from operational error, instrument failure and lightning. Post-accident recovery of damaged fuel may require such systems as overpacks or dry- transfer systems because ready retrieval of the fuel is required only for normal and off-normal conditions.

In addition, NUREG-1536 lists several natural phenomena events that should be evaluated, including flood, tornado, earthquake, burial under debris, lightning strike, seiche,[93] tsunami, and hurricane. The longer the expected period of dry storage, generally, the more severe the natural event loading will be that should be employed in analysis. Effects of each accident should be analyzed for its possible consequences on dry-storage systems. At one dry-storage site, NRC determined the following were the key accident initiating events that should be evaluated as most likely to breach a multi-purpose container: seismic events, aircraft strike, meteorite strike, vent blockage, and fire.[84]

It is possible that an off-normal or accident condition could lead to a decision being made to repackage the fuel assemblies that reside in a dry-storage container. This transfer may take place "wet" in a pool. If dried rods are returned to the reactor pool and if they are still relatively hot, when they are lowered into the water, the rods will experience a rapid quench cool-down. This may induce high stresses in the cladding and possible fuel-rod failure. It also is unknown what effects rewetting may have on the fuel and its future degradation.

A serious accident may lead to several unacceptable situations. A breach of the main canister may allow the release of radioactive material. Fuel previously oxidized to the U_3O_8 form "is a fine powder that spalls from the fuel surface. The release of fines and/or fuel relocations from split cladding must be evaluated if U_3O_8 formation is suspected. The extent of oxidation of irradiated UO_2 is a time and temperature-dependent phenomenon."[48]

A breach of the main canister will let helium out and air in. If a breach allows water or humid air to enter the container, then the likelihood of degradation of the cladding, canister, cask, and other components increases significantly. Possible effects of the humid atmosphere include radiolysis of the moisture to create highly oxidizing radicals, corrosion of the cladding and cask components, and enhanced hydrogen ingress into the cladding. Stress corrosion cracking, galvanic corrosion, pitting and other forms of localized corrosion may also occur. Given a breach, the entire cladding and system components will need to be inspected.

Severe accidents during transportation of CSNF may induce large impact stresses that could heavily damage aged fuel-rods and fuel assembly containment supports. Again, this possibility needs to be studied in more detail since loss of fuel-rod confinement and assembly containment could make subsequent storage and transport of damaged CSNF unsafe.

4.3.3. Physical Security Events

In the context of this report, *security* refers to measures to protect nuclear used-fuel storage facilities and transportation canisters against theft, sabotage, or terrorist attacks. Since the attacks of September 11, 2001, there has been an increased awareness of the potential security threats that could be directed against stored or transported used-fuel in an effort to cause grave harm to the public and to cause fear, social disruption, and economic loss. Thus, used fuel storage and transport may present an attractive terrorist target. Discussion of the various scenarios that terrorists could employ to attack a dry-storage site or transport vehicle is beyond the scope of this document. However, it should be noted that the ability to perform an adequate security analysis to protect against such attacks that could cause a significant release of radioactive material to the environment is a timely and important problem. Further, whether a breach of a dry cask or canister occurs due to a security event, accident, or natural event, there is clearly overlap between safety and security consequences and analysis.

The major difference between safety and security events is that while many initiating safety events are accidental or random, a physical security initiating event is intentional and targeted, and so potential security threats can change with increased dedication of resources or availability of new technology (such as shaped charges) and execution technique. There have been recent[94] and are ongoing studies by other government agencies to evaluate terrorist threats. Whereas a relatively sufficient experience base allows for quantification of safety risk probabilities and consequences, this is not the case for terrorist events, which are much more uncertain (now but especially into the future). Also unsettled is whether to rely on analyses using reference scenarios that are based on characteristics of previous attacks and expert judgment (mean-value best estimate release), or to consider maximum-credible consequence event scenarios (largest credible release).

NRC issued a "Draft Technical Basis for Rulemaking Revising Security Requirements for Facilities Storing SNF and HLW" in December 2009.[95] The schedule calls for the technical basis for the rule to be completed by May 2010, the proposed NRC rule to be issued by fall 2011, the final rule to be published by spring 2013, and licensee implementation of the new rule to be completed by the end of 2014. This rulemaking applies to dry cask storage only and not reactor pools. Objectives of the new rulemaking are threefold:

1. Update the Independent Spent Fuel Storage Installation (ISFSI) security regulations to improve the consistency and clarity of 10 CFR 72 regulations for both types of ISFSI licensees (i.e., general and specific) and to reflect current Commission thinking on security requirements;
2. Make generically applicable requirements similar to those imposed on ISFSI licensees by the post-9/11 security orders; and
3. Use a risk-informed and performance-based structure in updating the ISFSI and Monitored Retrievable Storage Installation (MRS) security regulations.

The most significant change is the new requirement for utilities to carry out a risk-informed and performance-based approach that calculates dose based on specific security scenarios at every ISFSI site. A 0.05-Sievert (5-Rem) dose limit requirement will apply at the controlled area boundary of an ISFSI. If the calculated dose exceeds this limit, licensees are to use an iterative design process in changing security boundary dimensions, or other engineered security features to decrease dose. Failure to meet the dose requirement will require a shift to a "denial protection strategy" to prevent the initiating terrorist event from occurring. The security scenarios include both land and waterborne based attacks and would require protection both of the CSNF and HLW, alarm stations, and security personal defensive positions.

One reason for the new rulemaking is that new NRC threat analyses were carried out against four cask designs (results classified). Formerly, it was assumed that dry casks were sufficiently robust to resist terrorist attacks, but employing new post 9-11 threats resulted in a risk of breach and possible radioactive releases. A second reason is the likelihood that CSNF and HLW will remain in dry storage longer than originally anticipated.

The dose limits will apply to all types of casks and their locations. The requirement is for risk calculations to be carried out at each individual site to quantify the risk to the public. To carry out the calculation, various security scenarios will need to be applied to a particular cask, in a particular location, with an assumed fuel burnup. The risk-informed calculation will result in the possibility of a breached cask with particular radiological release consequences. The performance-based part of the requirement refers to whether the plant will be able to tolerate the calculated consequences or whether other options may be required.

Over the period of extended dry storage, the amount of radiation from the stored used fuel decreases and, after some time, may no longer pose a deterrent to individuals approaching CSNF. However, from Figure 28, it can be seen that the CSNF remains quite radioactive over the first 100 years of storage.

4.3.4. Summary of Significant Storage Events Considered

Normal dry-storage handling and storage events under expected environmental conditions should not present a problem in satisfying evaluation criteria for many years. In the very long-term, however, an atmospheric corrosive environment could degrade the metal canister and other system elements, including handling components and overpack concrete, depending on the external environment (these will be dealt with in the next subsection). To model corrosion degradation, the future canister temperature needs to be known so the time when condensation and chemisorption on the canister surfaces can be predicted.

Most important to satisfying the evaluation criteria are the plausibility of certain severe accidents that may occur during dry storage. These include extreme off-normal and accident events (airplane impact, high drop), extreme natural events (earthquake), and extreme terrorist events. Any one of these could result in impact stresses that cause minor and possibly gross breaches of fuel rods and significant damage to fuel assembly containment supports. Research is needed to study the effects of and risks posed by potential severe accidents on long-term storage of aging systems and fuels.

4.4. Active Degradation Processes Affecting Features

Identifying most technical concerns associated with long- and very-long term dry storage is straight-forward, but assessing their relative importance is difficult because of the lack of data and analyses. In general, however, most of the environmental conditions experienced by the fuel, cladding, and dry-storage systems are largely known and understood. Some of them are site-specific, such as climatic conditions and environmental chemistry, while other conditions are similar for all CSNF.

Although radiation and temperature conditions are much more extreme in the nuclear reactor, some degradation processes will be active during the period of extended storage and these need to be evaluated. During the first 20 years of storage, the properties of the fuel, cladding, and dry-storage system components may change due to conditions such as elevated temperatures, presence of moisture and effects of radiation. During normal storage in an inert atmosphere, there is the potential for the cladding's mechanical properties to change because of temperature and interaction with fuel and cask materials. These potentialities are explored in the next three subsections of this report.

4.4.1. Behavior of Used Fuel-Pellets

The prediction of the containment, safe retrievability, and transport of used fuel are major safety topics of importance. A variety of mechanisms have the potential to degrade the fuel pellets. However, if the fuel pellets are maintained within an inert He environment, the most significant fuel-degradation mechanism (oxidation) is largely inactivated (residual water may result in some oxidation). Some of the other potential fuel-degradation mechanisms will require additional research to extrapolate their effects beyond current knowledge of dry storage.

a. **Fuel-pellet oxidation**. Uranium dioxide (UO_2) fuel is thermodynamically unstable in air at high temperatures and irradiated UO_2 exposed to an oxidizing atmosphere eventually will oxidize to U_3O_8. Thus, for the first decades of dry storage, ensuring the integrity of the fuel cladding by maintaining helium cover gas to prevent air or water from accessing the fuel pellets is important. Contact between fuel rods and an oxygen-containing atmosphere could result in UO_2 exposure to an oxidizing agent (oxygen) if a tiny cladding breach, such as a pinhole or a hairline crack, develops. Pacific Northwest National Laboratory (PNNL) produced a 2008 report that summarized knowledge of the oxidation of CSNF in air.[42] "The oxidation of UO_2 to U_3O_8 is a function of the time at temperature, initial grain size, burnup, and oxygen availability." However, PNNL found that temperature is the most important parameter followed by grain size when determining the possibility and extent of oxidation. Oxidation rates for irradiated UO_2 fuels were significantly faster than for unirradiated UO_2.[47]

   As mentioned previously, each fuel pellet is commonly split into multiple fragments in the reactor. Typically, oxidation of a used fuel-pellet exposed to air would begin at exposed surfaces (i.e., the pellet exterior surface and surfaces along pellet fractures), then tend to progress along grain boundaries, and finally spread homogeneously within the fuel fragments.[96] The existence of tunnels in the matrix would allow local spreading of the oxidation front.

   For almost all fuels of interest, fuel pellet (uranium dioxide) oxidation has been shown to occur as a two-step reaction, or three-steps if the plateau or incubation time is explicitly considered. First, the oxidation of the cubic UO_2 phase creates another cubic phase denoted by U_4O_{9+x} or $UO_{2.4}$. Because this oxidation is accompanied by a slight contraction of the fuel, no swelling and no additional cladding unzipping is expected to occur. For fuels with burnups greater than about 15 GWd/MTU, as long as "…the temperature of the fuel can be maintained at or below about 240ºC, then during the 720 hours [30 days] of exposure, only oxidation to $UO_{2.4}$ will occur."[97] The $UO_{2.4}$ oxidation product is not a fine powder available for airborne release.

Some low-burnup samples (possibly with hydrated phases present) immediately start converting to the U_3O_8 phase, but dry CSNF samples show a plateau or incubation time at $UO_{2.4}$ before final oxidation to U_3O_8 occurs. The plateau behavior has been shown to be strongly temperature, grain-size, and burnup dependent.[98] Table 6 summarizes the plateau or incubation time needed at 8 and 27 GWd/MTU while maintaining fuel at various temperatures. Higher burnup fuel has longer plateau times than low burnup fuel, especially at lower temperatures. Brady in a Yucca Mountain analysis and modeling report presented similar and more conclusive results for plateau times as a function of burnup and temperature (see Table 7) using assumptions that were conservative.[99] Similar dependencies are found for the final oxidation step.

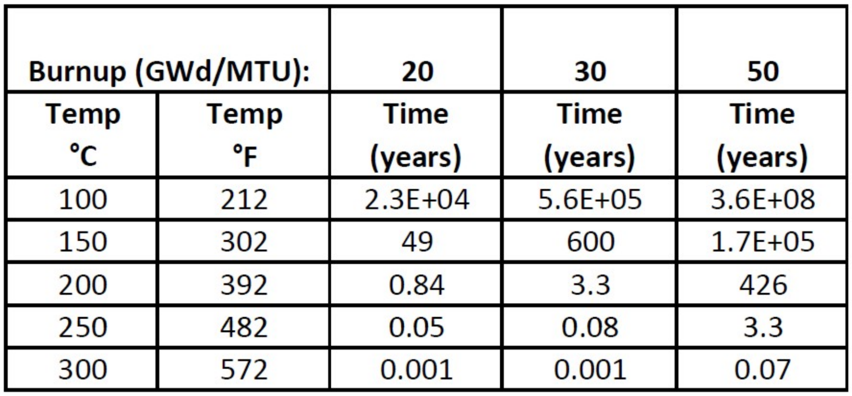
Table 6. Plateau or Incubation Time of CSNF as a Function of Temperature and Fuel Type[100]

		CANDU	**PWR/BWR**	**PWR/BWR**
Burnup (GWd/MTU):		**8**	**27**	**59-66**
Temp ºC	**Temp ºF**	**Time**	**Time**	
100	212	28 y	1868 y	
150	302	1.5 y	28 y	
200	392	58 d	1.0 y	
250	482	9 d	26 d	
300	572	2.0 d	2.8 d	
350	662	0.5 d	0.4 d	none*
400	752			none*
450	842			all**
500	932			all**

* After 30 days had only oxidized to $UO_{2.4}$ phase
** After 30 days all of sample was oxidized to U_3O_8 powder

After an individual grain has fully oxidized to U_4O_{9+x} or $UO_{2.4}$, it then oxidizes over time to U_3O_8.[42] This transformation to U_3O_8 destroys the ceramic structure, reducing the material to a grain-size very fine powder. As mentioned previously, temperatures higher than 240ºC are required for oxidizing fuel over a period of days to weeks (see Table8). UO_2 , if exposed to air, then undergoes a transformation sequence leading to orthorhombic U_3O_8, which has a 31 to 36 percent greater volume. This dilational (volume increase) process, in this instance due to chemical oxidation, is called "bulking" and can induce mechanical stress that is sufficient as to enlarge an initially small breach. Continued oxidation and bulking, which can reach more than 300 percent, will exert additional stress and lead to rupture of the fuel cladding.[101]

Table 7. Plateau Time (yr) of CSNF as a Function of Temperature and Burnup[99]



Burnup (GWd/MTU):		20	30	50
Temp °C	Temp °F	Time (years)	Time (years)	Time (years)
100	212	2.3E+04	5.6E+05	3.6E+08
150	302	49	600	1.7E+05
200	392	0.84	3.3	426
250	482	0.05	0.08	3.3
300	572	0.001	0.001	0.07

Time to reach a fuel pellet/cladding gap of zero.

Source: B. Hanson, "Clad Degradation – Dry Unzipping,"
Analysis & Model Report, CRWMS M&O (2000), see Table 15

The oxidation behavior depends on certain fuel characteristics, in particular, the temperature and irradiation history (burnup and linear power).[102] Lower temperatures lead to less oxygen diffusion in the fuel matrix (resulting in less fuel oxidation). Higher-burnup fuel contains increased fission products and actinides that hinder oxygen migration and help stabilize the intermediate oxidation phase (U_4O_{9+x}) – so higher-burnup fuels take longer to oxidize to U_3O_8. Fuel exposed to higher linear power has been correlated to "tunnels enabling the fast spreading of oxygen in the used-fuel fragments."[102] Accordingly, lower-linear-power fuel results in fewer tunnels and oxygen-diffusion pathways. One researcher found that complete oxidation to U_3O_8 occurred within about 10 hours at temperatures approaching 400ºC for a fuel with a 65 GWd/MTU burnup. Detailed reviews of fuel-pellet oxidation testing "emphasize that unirradiated UO_2 and CSNF oxidize via different mechanisms and at different rates."[98] Oxidation rate data obtained on unirradiated samples may not be directly applicable to long-term dry storage of CSNF. Furthermore, the presence of moist air at high relative humidity may slightly increase the oxidation rate at low temperatures, but research is needed to quantify this effect.[98]

The U_3O_8 grains are "…a fine powder that spalls from the fuel surface. The release of fines and/or fuel relocations from split cladding must be evaluated if U_3O_8 formation is suspected. The extent of oxidation of irradiated UO_2 is a time and temperature-dependent phenomenon."[48] Because fuel rods are under significant internal pressure, any accidental cladding breach would be accompanied by a release of pressure and gas outflow from the rod through the breach. A small breach could provide a sustained, low-volumetric gas outflow; a large breach would tend more toward a sudden, relatively high volumetric flow rate.

Once pellets are out of the reactor, continuing α-decay radioactivity causes limited new irradiation damage in the UO_2 matrix (solid swelling via displacement cascades) and slight gaseous swelling induced by the production of helium and other fission gases. In general, however, fuel pellet-fragments in wet storage are assumed not to experience significant degradation, primarily due to the helium atmosphere and tightly controlled temperature and water chemistry (see subsection 4.1.2.2). Consequently, during the period *of wet storage*, there is an insignificant risk of fuel pellets oxidizing and causing a gross cladding breach by unzipping of the cladding. *Minor oxidation could occur at the pellet surface of failed fuel due to the formation of radiolysis products.*

One of the more mechanically stressful steps in dry storage is the drying process. This process has the potential to affect cladding more than fuel pellets, so it is discussed further in subsection 4.1.2.3. Under drying conditions, the physical state of fractured fuel-pellets continues to evolve because of the cycles to high temperature changes, continuing radiation, and the various physical and chemical heterogeneity and molecular concentration gradients present. Temperature limits are imposed to minimize these physical changes. These limits are for ensuring that the cladding is not likely to fail over the near term and that the cladding continues to contain an inert helium atmosphere.

The oxidation of UO_2 to U_3O_8 involves an intermediate oxidation product, as described previously, and is a function of several factors, but temperature is the most important parameter. Table 8 summarizes the time needed for complete oxidation while maintaining fuel of various burnups at particular temperatures. The temperature conditions expected over the first 20 years of dry storage, coupled with this experimental data, led, in part, to the decision to store fuel in an inert atmosphere cover gas.[24]

Table 8. Time for Complete Oxidation of CSNF to U_3O_8 as a Function of Temperature and Burnup

Burnup (GWd/MTU):		20	30	50
Temp °C	Temp °F	Time (years)	Time (years)	Time (years)
100	212	3.5E+06	8.9E+07	5.6E+10
150	302	5440	9.3E+04	2.8E+07
200	392	33	417	6.8E+04
250	482	0.5	5	522
300	572	0.02	0.14	9

Source: B. Hanson, "Clad Degradation – Dry Unzipping," Analysis & Model Report, CRWMS M&O (2000), see Table 12

Because fuel rods are under significant internal pressure, any accidental cladding breach would be accompanied by a release of pressure and gas outflow from the rod through the breach. A small breach could provide a sustained, low volumetric gas outflow, while a large breach would tend more toward a sudden, relatively high volumetric flow rate. Because oxidation typically occurs first along the fuel grain boundaries, there is usually a release of fission gases, such as ^{85}Kr, to the container quite early.

If the temperatures are hot enough during the early years of dry storage, and if there is a source of oxygen, from Tables 6 to 8, it is possible that a portion of the used fuel could oxidize in the first couple of decades. For avoiding fuel oxidation during dry storage, three mitigation options have been identified by NRC and vendors, and discussed in NRC guidance documents. The three options are to maintain the fuel rods in an inert atmosphere (helium gas in the fuel pin and in the canister or cask), ensure the absence of cladding breaches, and keep the time the fuel is undergoing active oxidation shorter than the time required to initiate rod splitting. Low-burnup (about 30-35 GWd/MTU) Surry PWR fuel rods were kept in dry storage for 15 years in a Castor V/21 cask; there was no cladding breach so there was no noticeable fuel oxidation of the extensively cracked fuel.[103]

An example of the potential hazard posed by transporting or unloading failed oxidized fuel occurred in July 1980.[104] During operation in the Connecticut Yankee reactor, one or more rods failed and resulted in a fuel assembly being classified as damaged (more than pinhole leaks or hairline cracks in the cladding). The assembly was withdrawn and it was desired to send the damaged assembly to Battelle Memorial West to evaluate the cause of the cladding failure. The assembly was loaded into a NAC-1E transport cask, filled only with air. Transport to Battelle in Ohio occurred without incident. Subsequent evaluation determined that, during transport, the fuel reached a temperature of about 430°C (800°F). At this temperature, the exposed fuel meat likely scavenged oxygen from the container air, oxidized to U_3O_8, and this process led to further cladding damage by unzipping, fed by more fuel and being exposed to air. As a result, the transport container was internally contaminated with oxidized used-fuel powder and fragments.

The bolted cask lid was removed in the Battelle unloading pool, approximately 23 feet underwater. It is believed that a hot column of water rose to the pool surface carrying “a significant (very large) quantity of fuel particulate. A significant portion of that particulate became airborne, which severely contaminated the unloading pool area.” It is not fully understood how the oxidized-fuel particulates became airborne and contaminated the area. Because of this incident, new transportation regulations require that transport casks be filled with helium instead of air, to prevent further fuel oxidation and unzipping. Further research is needed to better understand the behavior of particulate fuel in water and air, and how to safely handle oxidized fuel in wet and dry environments.

b. **Release of fission-product gas from used fuel-pellets to the fuel-rod gas space**. During irradiation of fuel in a reactor, fission products are created, some being noble gases such as xenon and krypton, which are insoluble in the fuel. Uranium-dioxide fuel-pellets are manufactured with a mostly closed porosity of about 5-7 percent to help contain gaseous fission products. However, some of the fission-product gases are still released from fuel pellets, which increases the fuel-rod internal gas pressure (within the cladding). In high-burnup fuels, generation of fission gas has the physical effect of increasing porosity to 20 percent or more at the pellet rim.

The phenomenon of fission-gas release has been studied for decades and several release mechanisms have been identified. The primary release mechanisms are fuel cracking, recoil and knockout releases, and diffusion of the gases out of the fuel.[105] Some gas release occurs during irradiation because of fission-fragment knockout of near-surface fission gas. Recoil release occurs right after fission near the external surface when fission gases with high kinetic energy escape. Knockout releases occur when fission fragments pass through a specimen and either directly collide with gases or knock out a small physical portion of the uranium oxide surface, releasing some fission gases. During storage, the dominant fission-gas release mechanism is diffusion.[106]

Most of the diffusion fission-gas release is accomplished in two steps[107,108] (see Figure 32), both of which are heavily dependent on fuel burnup and temperature. First, gas diffuses from the fuel matrix to the grain boundaries, and may coalesce to form small bubbles. Next, these bubbles coalesce along the grain boundaries forming larger bubbles that migrate to the pellet surface. Some gas diffuses as single gas atoms. From the French perspective, the "present state of knowledge suggests that in a first step, helium should accumulate in intragranular bubbles possibly causing microcracks. This phenomenon should accelerate the helium release to grain boundaries and by increasing the preexisting fission gas bubble pressure lead to opening of grain boundaries as observed in old PuO_2 pellets [after 30 years]. The long term fate of the helium in the used fuel is the major open question concerning the behavior of used fuel in a closed system."[109]

No theory seems adequate for accurately predicting gas release from cracked fuel-pellets. Gas does not diffuse back into the fuel, and its production does not slow the diffusion out of the fuel. Subsequent damage to the pellet matrix will allow further release of trapped fission gas in the overpressurized pores.[110] One type of damage that results in the release of fission gas is oxidation of the fuel fragments. "[I]t has been shown recently that the release of intergranular fission gas is proportional to the progression of the oxidation front along the grain boundaries for UOX fuels."[111]

The diffusion process continues during wet and dry storage to add marginally to the inventory of gaseous products (Kr and Xe) in the fuel-rod gas space. At wet- and dry-storage temperatures, pure thermal diffusion is the major active mechanism for releasing fission-gas products, but at low temperatures, there is essentially no release of gases at all by thermal diffusion, even over extended periods of time. Thus, as the fuel cools and the α-activity declines, diffusion rates are expected to decrease.[109] However, the fission gases already in the plenum could rapidly be released with a cladding failure. Such a depressurization of the fuel rod could lead to further release of fission-gas products from the pellets. In addition, according to the ideal gas law, the plenum gas pressure will decrease as the internal fuel-rod temperature decreases. Some experts feel the expected small amount of gas diffusion in stored fuels needs to be verified.[109]

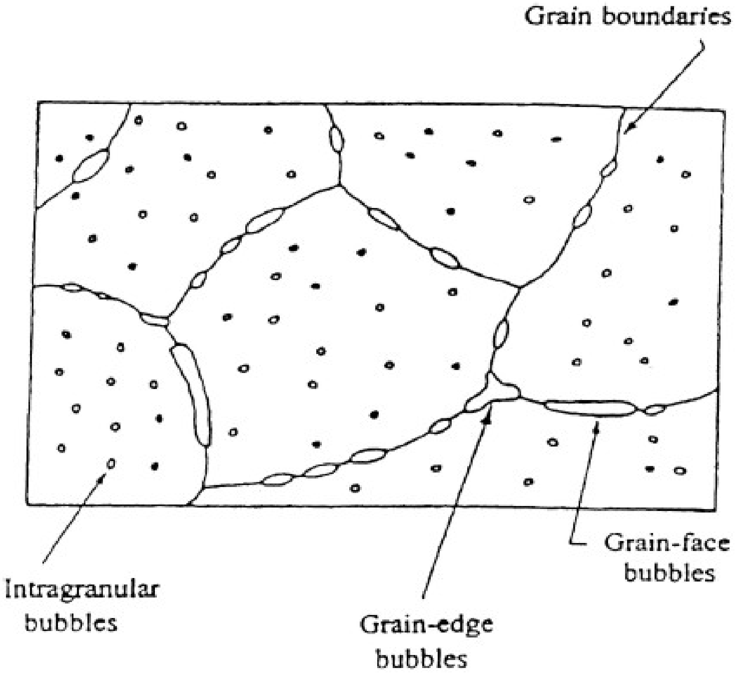


Figure 32. Schematic Cross Section of Granular Fuel-Pellet with Gas Products[112]

For the case of low-burnup (about 30-35 GWd/MTU) Surry PWR fuel rods kept in dry storage for 15 years in a Castor V/21 cask, the central region of the fuel pellet (seen in cross section in Figure 32) retained a greater amount of fission-gas bubbles, implying limited diffusion. In addition, calculations indicate that the measured fission-gas release occurred in-reactor and not during storage and that any further dry-storage release amounts "would have an insignificant effect on internal pressure and cladding stress."[113] However, other researchers believe that release of fission gases from the fuel may increase gas pressure within the cladding, which increases cladding hoop stresses. Japanese researchers assumed a "conservative estimation" of an internal fuel-rod pressure of 8 MPa (1160 psi) at room temperature for PWR used fuel-rods.[114] Diffusion of fission gasses out of fuel pellets is slow at low temperatures, but over very-long-term storage timeframe, could this phenomenon become important? Research may be needed to confirm that fission gas diffusion will not cause significant problems.

In an instance of a waste package failure and cladding breach (not an explosion), some volatile radionuclides, present in the plenum, could be rapidly released. The radionuclides of most interest include the Xe and Kr fission gases and the volatile elements I, Cs, and Cl. Longer-term release of radionuclides is not considered a likely problem because the release should be detected and recanistered.

c. **Helium gas microcracking.** Helium gas is produced within ceramic fuel-pellets over time because of the α-decay of actinides (α particles, helium nuclei, are released, then combine with electrons to create helium atoms). The helium atoms are accommodated in the ceramic interstitially, trapped at internal defects, aggregate to form bubbles, or are released at the ceramic surface. The α self-irradiation process continues during used-fuel storage. The helium concentrations are highest at the center of the fuel pellets and drop off near the fuel-pellet boundary. Fuels containing greater amounts of plutonium (e.g., MOX fuels), produce significantly greater amounts of helium than traditional UO_2 fuel.

Generation of helium gas can lead to intergranular and intragranular gas bubbles that may cause swelling, matrix microcracking, and release of trapped gases similar to the effects of fission-gas release. Once the fuel has cooled, diffusion of helium essentially stops and the helium remains occluded in the fuel matrix or trapped at the grain boundaries. More research is needed for determining whether further production of helium leads to microcracking liable to induce grain decohesion over the period of long-term storage.[109,115,116]

Microcracking may be a more prevalent problem in MOX fuels. Present knowledge indicates that this process occurs slowly. Helium gas release to grain boundaries can "lead to opening of grain boundaries." Studies of He solubility, diffusion, trapping, and release in ceramics are needed. Damage to UO_2 fuel-pellets in the form of "grain subdivision or polygonization of high-burnup UO_2 fuel (the so-called rim-effect)… may be gas-driven."[117]

d. **Effects of radiation damage to ceramic-fuel matrix during storage.** Beta and gamma activity predominates over alpha activity in used fuel at the time it is unloaded from the reactor. The principal sources of radiation in used fuel are the β-decay of the fission products (e.g., ^{137}Cs and ^{90}Sr) and α-decay of the actinide elements (e.g., U, Np, Pu, Am, and Cm). Beta-decay produces energetic β-particles, very-low-energy recoil nuclei, and γ-rays, whereas α-decay produces energetic α-particles, energetic recoil nuclei, and some γ-rays. There also are minor contributions to the radiation field from spontaneous fission of some of the actinides and from (*α, n*) reactions, but they do not contribute significantly to the overall radiation effects. "In general, β-decay is the primary source of radiation during the first 500 years of storage, as it originates from the shorter-lived fission products."[118]

Radiation-induced physical and chemical changes in the ceramic used-fuel waste-form can occur both at the atomic and macroscopic levels, and some effects of radiation will accumulate over time. Radiation effects on crystalline materials can be broken down into two categories: the transfer of energy to electrons (ionization and electronic excitations) and the transfer of energy to atomic nuclei, primarily by ballistic processes involving elastic collisions. For β-particles and γ-radiation, the energy transfer is dominated by ionization processes. For ions, such as α-particles and recoil nuclei, interactions involve ionization processes and elastic collisions. In addition to the transfer of energy, the particles emitted through radioactive decay can themselves, in some cases, have a significant chemical effect on the nuclear waste material as a result of their deposition and accommodation in the structure.[119]

Energy loss by β-particles and α-particles is predominantly through electrostatic force interactions (Coulombic interactions). The interaction of γ-rays with matter is primarily through the photoelectric effect, Compton scattering, and electron-positron pair production. The high rate of energy absorption through ionization and electronic excitation from β-decay can result in self-heating and can result in covalent and ionic bond rupture, charged defects, enhanced self-ion and defect diffusion, localized electronic excitations, and sometimes, permanent defects from radiolysis (destruction of chemical bonds).[120] Radiolysis of any water present (mainly due to α-particles but also gamma radiation) involves water breaking down into hydrogen peroxide, hydrogen radicals, and assorted oxygen compounds such as ozone. Ballistic processes cause direct atomic displacements through elastic scattering collisions and are responsible for the atomic-scale rearrangement of the structure. Beta particles, because of their low mass, do not cause significant atomic displacements; most displacements are caused by the high energies of the α-particle and α-recoil nucleus produced in a α-decay event. These particles are released in opposite directions and produce distinct damage regions separated by several microns.[121]

The irradiation affects on the physical and mechanical properties of pellet material can be stated as: irradiated ceramics lose density, swell, store energy, and soften, resulting in decreasing linear elastic moduli, decreases in hardness, and increases in fracture toughness. Radiation-induced swelling can result from the accumulation of point defects in the crystalline structure, solid-state phase transformations (e.g., amorphization), and the evolution of microstructural defects (e.g., gas bubbles, voids, dislocations, and microcracks). The net result of the decreases in hardness and elastic modulus and increases in fracture toughness is a reduced brittleness and enhanced resistance against crack propagation.[122]

A detailed understanding of the radiation-damage accumulation over time is important to predicting the long- and very-long-term behavior of damaged materials. However, the physical effects of in-reactor radiation likely dwarf any further cumulative effects over extended storage. “In the case of spent nuclear fuel, the radiation dose due to the in-reactor neutron irradiation is already substantial, and additional damage accumulation during disposal is not anticipated to be significant.”[123] Therefore, as fuel pellets cool in extended storage under the radiation environment described in this subsection, no significant further damage is expected (more cracking, etc.). Additional research on this mechanism is not considered necessary.

e. **Combinations of mechanisms.** Research generally focuses on one or the other of the fuel-degradation mechanisms discussed above and does not take account of mechanisms acting simultaneously or in a coupled manner. For developing long-term technical bases, the contribution of simultaneous combinations of mechanisms should be addressed by new research and continued periodic characterization of representative fuel types that are stored.

Summary: degradation behavior of used fuel-pellets. The design of dry-storage systems rely on two independent barriers to prevent the oxidation of used fuel-pellets: helium in the fuel rod and helium in the canister. Helium is an important part of the safety system of dry cask-storage systems and its presence needs to be verified and ensured for preventing activation of extended dry storage degradation mechanisms. Currently, neither pressurized helium in fuel rods nor in welded canisters is monitored. Research into monitoring the presence of helium within welded casks, preferably nondestructively, would be helpful. Helium detection may not be of the highest priority if, after some years, dry-storage fuel temperatures are low enough to prevent significant oxidation to U_3O_8, as current research appears to indicate (but the potential for low-temperature oxidation of fuel needs to be better understood). If the helium cover gas escapes the cask system, or for the case of a cask breach during accidents, it would be helpful to further characterize the oxidation behavior of various types of fuel in air, with temperature. If even a small proportion of CSNF fuel rods are discovered to contain oxidized fuel, research is needed to better understand how the powdered fuel may be dispersed and spread upon opening a canister under water or in air. Fission-product gases are hazardous (radioactive) and may also contribute to overpressure of fuel rods. Validating the modeling that has been done and extending it to high-burnup fuel would be useful. Helium gas microcracking appears to damage fuel pellets and is especially active in MOX fuels, but not much data are available for understanding this phenomenon and whether it triggers more release of fission gas or other problems.

4.4.2. Behavior of Fuel Cladding and Cladding Integrity

Because a small number of used-fuel assemblies experience cladding failures during their three to four years in the reactor, it is possible that some intact fuel-rods experience damage and in situ stress conditions that may bring them close to failure. Once fuel rods are removed from reactor conditions, the cladding temperatures, internal pressures, and mechanical stresses are decreased significantly. Only during the drying process and subsequent accident conditions do temperatures and stresses spike again. Given the typical initial state of cladding and the potential for continued degradation, additional used-fuel cladding failures may occur during dry storage, handling and transport. The frequency of fuel rods that are expected to fail during dry storage and transportation may be analyzed using different assumptions, but the predictions of these simulations can be validated only by performance testing using representative sampling and inspection of stored rods under actual conditions over years.

A variety of important mechanisms may have the potential to degrade the cladding barrier over the long and very long-term (some shown in Figure 33). The conditions that govern these degradation mechanisms are strongly dependent on the initial physical states, temperature, and in situ mechanical stress state of the cladding when fuel rods are first placed in dry storage. One 1986 short-term German study, summarizing four years of laboratory research on dry storage of used fuel with burnups of 16 to about 40 GWd/MTU, concluded, “Fuel behaviour indicates that dry storage under inert gases is not expected to cause any cladding failures over the interim storage period [a decade or two] as long as the dry storage temperature is limited to 450ºC.”[124] A year 2000 EPRI report concluded, “…the general technical consensus is that creep is the governing mechanism for spent fuel in long-term dry storage” after considering stress corrosion cracking, delayed hydride cracking and creep rupture as possible degradation mechanisms.[125] An EPRI 2007 cladding performance synthesis report, assuming about 40 years of dry storage, concluded, “…damage to materials and structures of high-burnup spent fuel systems under prescribed regulatory conditions of dry storage and transportation is tolerably small, with little or no impact on spent fuel operation management” and that postulated failure mechanisms “…are either inactive or lack sufficient driving forces to cause cladding failure.”[63] These conclusions may still be true after a much longer extended period of dry storage, but new research is discussed below that may indicate that hydrogen embrittlement and delayed hydride cracking could be the governing mechanism in long-term dry storage, not creep rupture. Low temperature, low pressure degradation mechanisms could be important over a period of extended storage. There also is some uncertainty regarding the role of residual water that is left on cladding and in the canister after drying, concerning whether it could lead to significant degradation. Several potential cladding-degradation mechanisms will require additional research and direct observations over time to understand their significance and confirm assumptions over several decades and longer periods of extended dry storage.

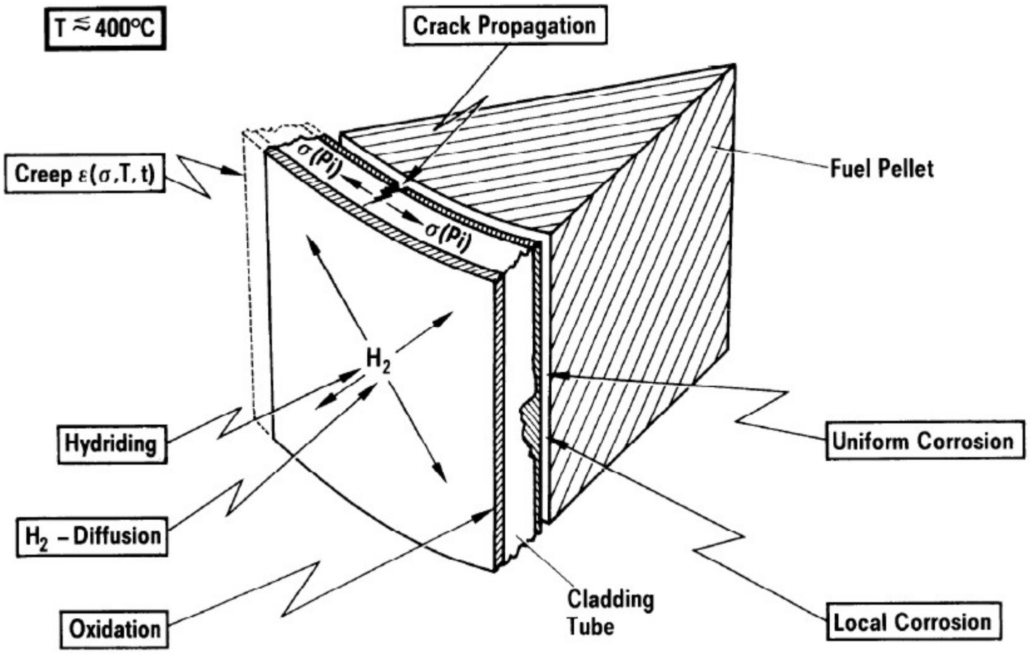


Figure 33. Mechanisms Affecting Used-Fuel Cladding Performance During Dry Storage[126]

a. **Tangential creep rupture.** Sufficient mechanical ductility of cladding is important to prevent ruptures. Creep is the progressive deformation of a material under stress. Unlike an operating nuclear reactor, the pressure within a fuel rod in storage is not countered by any positive external pressure; in fact a negative vacuum outside the rod during drying increases the differential pressure substantially. This internal pressure creates the tube hoop stresses that are primarily responsible for cladding-creep deformation. Whereas the creep rate can be defined mostly by levels of stress and temperature for a specific material, creep tearing or rupture depends on an additional parameter, such as strain rate or holding time, and therefore requires considerably greater experimental effort for its evaluation. Creep rupture of Zircaloy cladding during vacuum drying is one of two most likely failure mechanisms for used fuel-rods.[127,128] When the canister vacuum is removed and backfilled with helium, the differential pressure is much less and the creep rate decreases accordingly, decreasing the risk of rupture (at least in the near term).

Creep at constant stress occurs in three approximate stages. The primary stage has rapid deformation and a decrease in creep rate over time. The secondary stage has a constant creep rate, and the tertiary stage has a rapid creep-rate increase with time until fracture occurs. The two physical conditions influencing creep are the hoop stress and the temperature. Over long storage times, both the temperature and the internal cladding pressure decrease and thus the creep strain rate drops off to be insignificant and not all of the above stages may be seen. "However, these positive internal pressures could increase over the long-term by potential release of helium and fission gases to the void volume of the fuel rod."[129]

Hoop stress in the cladding, which is due to the differential gas pressure (negative external vacuum and internal positive pressure) in the fuel rod, is the primary driving force for cladding creep, but creep also increases with temperature. The internal gas pressure is due primarily to the initial He fill gas and, to a lesser extent, the fission gas released from the fuel into the fuel-rod gas space.[130] Japanese researchers assumed a "conservative estimation" of an internal fuel-rod pressure of 8 MPa (1160 psi) at room temperature for PWR used fuel-rods, which they calculate will result in a cladding hoop stress of about 140 MPa.[114] Under typical internal gas pressures, strain is usually measurable while the cladding temperature is over 300ºC.

Failures involving creep usually involve a significant amount of plastic deformation of the cladding caused by the fuel-rod internal gas pressure arising under dry-storage conditions. Cladding creep is the critical "degradation and failure mechanism for setting maximum allowable storage temperature limits in dry inert storage."[131] The relatively high temperatures (during vacuum drying operations and subsequent dry storage),[132] differential pressures across the cladding, which "may vary between 11 – 12.4 MPa"[132], and corresponding hoop stress on the cladding will result in permanent creep deformation of the cladding over time. Figure 34 illustrates newer (2006) creep models and test results that take into account observations of partial annealing of the irradiation defects that may yield higher values of creep.[109] Accelerated creep tests at higher temperatures and/or pressures than actual conditions are sometimes used to predict creep rupture of cladding.

During the years in dry storage, the fuel rods will tend to decrease in temperature as the radiation level drops, and the internal rod pressure generally decreases because of creep (larger gas volume) and the decaying temperature (although gas release from the fuel could increase pressures somewhat). Consequently, over time the cladding hoop stress decreases, which will decrease the creep rate, the total creep, and potential for gross cladding rupture if a small breach occurs.

Guidelines for preventing creep. Experimental creep results have shown that Zircaloy cladding retains sufficient ductility if kept below 400ºC. To ensure the mechanical integrity of the cladding material, the following criteria should be met: (1) for all fuel burnups, the maximum calculated fuel-cladding temperature for normal operations of drying, transfer (short-term loading), and storage should not exceed 400ºC; and (2) for short-term off-normal and accident conditions, the maximum cladding temperature should not exceed 570ºC.[133,134,137]

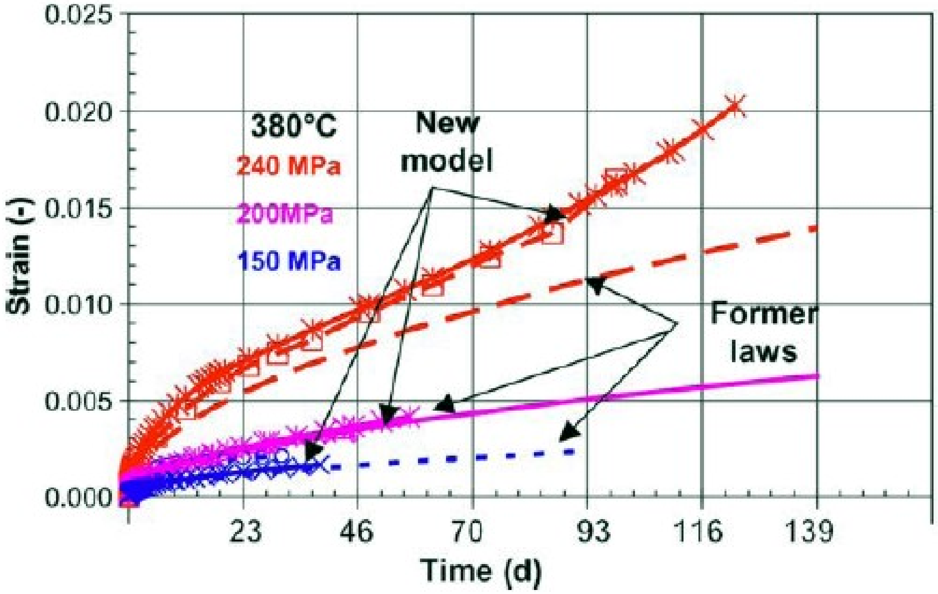


Figure 34. Comparison of Cladding Creep Models at Different Tube Hoop Stresses[109]

*Actual fuel-rod characterizations***.** For the case of low-burnup (about 30-35 GWd/MTU) Surry PWR fuel rods kept in dry storage for 15 years in a Castor V/21 cask, Einziger et al. determined that there was "very little" thermal cladding creep during the dry-storage period, and the "creep would not increase significantly during additional storage because of the low temperature."[135] German scientists calculated that high-burnup (55 GWd/tHM) LWR fuel assemblies would not be adversely affected by creep over a 40-year dry-storage period in a CASTOR V/19 cask.[136]

Argonne National Laboratory investigated segments of defueled high-burnup (about 67 GWd/MTU) Zircaloy-4 cladding from the PWR H.B. Robinson plant after pool storage and found "significant waterside corrosion and hydrogen uptake," as expected. They next performed thermal creep tests on the cladding and determined that the "higher hydrogen content in the Robinson material… appears to have no detrimental effect on creep behavior at the test temperature" which was 360-400ºC and 160-250 MPa hoop stress.[137]

Research is needed to better understand the creep rupture parameters that lead to maximum attainable strains under low-stress and low-temperature conditions. Most available creep and rupture experiments focus on the behavior of cladding over the first 10 years or so of storage when temperatures are higher. Research is needed that is focused on long-term creep behavior at low temperatures, using irradiated fuel, and under hydrogen-embrittled conditions to ensure that long-term conditions are truly benign. A potential source of cladding stress may be related to pellet swelling and creepdown. If there is no gap between pellet and cladding-wall in dry storage, as the fuel cools, could different coefficients of thermal expansion lead to higher cladding stresses and could this significantly affect creep rupture over the long-term? This phenomenon should also be examined to assess the long-term consequences.

The long-term experimental results using unirradiated fuel needs to be correlated with results from actual irradiated fuel. EPRI has been conducting important research of high-burnup fuel (results are proprietary), but more research is needed for studying how MOX fuel, aluminum cladding, and high-burnup fuel creep and rupture behavior may evolve. This includes research on specific new cladding materials over the long- and very long-term.

b. **Stress corrosion cracking (SCC).** In the reactor, cracks are generated within the cladding because of fuel-pellet-to-cladding thermomechanical interaction. These "micro," or incipient cracks, can be observed in an irregular pattern on the inner surface of fuel cladding.[60,138] Stress corrosion cracking involves the chemical corrosion of a crack tip driven by stresses in the cladding.

Besides hoop stress and cladding temperature, general creep crack-growth behavior of incipient cracks in cladding is known to be a function of gas content of the fuel, geometry and distribution of fuel pellets, irradiation, and cladding annealing history. Stress corrosion cracks on the interior surface of the cladding propagate by a combination of volatile-fission-product chemistry (e.g., iodine) and large tensile hoop stresses. The propagation of such cracks may result in effective increases in hoop stress, which would accelerate cladding-creep rupture.

"One of the causative agents of stress-corrosion cracking is known to be gaseous iodine, which is generated between the grain boundaries of the uranium dioxide during the fission reactions, hence called an iodine-induced stress-corrosion cracking."[139] Experimental testing of Zircaloy-4 cladding shows that highly corrosive iodine (at a concentration of 1.5×10^{-3} g/cm^2) initiates pits on the inner surface of the cladding at microcrack locations. The pits coalesce into a large pitting cluster and "pitting clusters play a role of nucleation sites" for an iodine-induced stress corrosion crack. These cracks initially proceed along grain boundaries, "and then propagate across the grains to leave a fracture pattern as a cleavage" or tearing. An iodine environment greatly enhances cracking.[140] However, other researchers feel "release of volatile fission products from the fuel pellets to the gap of an intact rod will be negligible."[48]

ASTM C 1562 – 2003 summarizes some of the science regarding iodine-induced SCC.[48] Essentially small amounts of iodine as a fission-gas product are released by the UO_2 fuel while in-reactor, but probably not enough to meet the threshold concentration needed for initiating SCC (greater than 5×10^{-6} g/cm^2). Below 400ºC, during storage, pure thermal diffusion is the only mechanism active to release new iodine gas, but at this temperature there is essentially no release of iodine at all by thermal diffusion, even over extended periods of time.

Guidelines for preventing SCC. German researchers limited hoop stress to 120 MPa and cladding strain to 1 percent to ensure cladding integrity over a storage period of 40 years or longer.[141] Japanese researchers found that the presence of cesium suppressed iodine

SCC and with a strain limit of 1 percent, concluded long-term, low-stress SCC degradation could be avoided.[114] Several laboratory programs have demonstrated that used fuel has significant creep capacity even after 15 years of dry cask storage. Therefore, it appears that a cladding strain limit of 1 percent will prevent cladding failure by SCC degradation.

c. **Hydrogen-Related Mechanisms and Effects**. During dry storage, the concentration and distribution of hydrogen in cladding can change, especially because of temperature gradients along fuel rods. This hydrogen may activate breach mechanisms such as hydride embrittlement and delayed hydride cracking (DHC). Cladding rupture due to formation of radial hydrides and then DHC during or after long-term storage is considered one of the most likely failure mechanisms of used fuel-rods.[128] Even a relatively small amount of radial hydrides has been found to be extremely detrimental to the integrity of cladding materials.[149]

Hydrogen absorption by zirconium alloys and subsequent precipitation as zirconium hydrides was discussed in subsection 4.1.2, “State of the Fuel Cladding.” The quantity of hydrogen absorbed, determined as the fraction of the total hydrogen generated, is known as the hydrogen pick-up fraction. Hydrogen (before precipitation) migrates in the cladding because of fuel-rod temperature and hydrogen-concentration gradients. During dry storage, the temperature gradient across the cladding [wall] is relatively small, leaving hydrogen concentration as the active driving force, so diffusion tends to even out the hydrogen distribution.[142] However, Figure 31 shows there is a relative large axial temperature gradient in cladding, which could lead to hydrogen migration to the cooler ends. This phenomenon needs to be studied to understand whether such migration and possible increased local hydride precipitation is significant for degradation of very-long-term cladding behavior. When the hydrogen concentration exceeds the solubility limit, hydrides start to form and grow. Higher-burnup fuels tend to have thicker oxide layers on the cladding, and this additional oxidation means that the cladding has had an opportunity to absorb more hydrogen.

Hydride reorientation involves the dissolving of pre-existing circumferentially oriented hydrides (see Figure 19) into the cladding and then the re-precipitation (under stress) of zirconium hydrides oriented perpendicular to the hoop stress (i.e., along the longitudinal direction of the cladding (radial hydrides)). Under particular combinations of cladding temperature, stress state, hydrogen concentration, cooling rate, and plastic creep, reorientation of all hydrides is possible during thermal cycling (repeated heat-up and cool-down cycles during the drying process). Higher burnup fuels may experience more hydride reorientation problems than low burnup fuels.[143]

During dry storage, gaseous hydrogen within the fuel rods is under a pressure that is relatively low in comparison to in-reactor or drying pressures. “Low-pressure gaseous hydrogen acting on materials for very long times may involve different mechanisms than those observed for high hydrogen fugacity. Therefore, high pressure hydrogen effects data should not be extrapolated to the low-pressure case of SNF storage.”[81] Specifically, researching the uptake of hydrogen under lower pressure and over long periods of time would be useful.

Hydrogen embrittlement. Hydride embrittlement is due to the formation of radial hydrides sufficient to cause detrimental effects on mechanical properties of the cladding, including tensile ductility, fracture toughness and ultimate fracture strength. In particular, as the proportion of hydrides with radial orientation increases, the effects of cladding hoop stress on cladding ductility become more significant. The hydrogen embrittlement mechanism is linked to decreasing fuel temperature and "must be considered during extended storage where the temperature is low."[143] Because higher levels of embrittlement can lead to fuel-rod failure as a result of the stresses endured during assembly unloading and reloading, or due to the stresses that fuel rods experience during accident scenarios associated with storage, handling or transport, minimizing hydrogen embrittlement is important.

Radial hydrides can degrade the mechanical properties of the cladding, particularly ductility. Metallographic examinations reveal that the embrittlement is caused by microstructural changes in the cladding, including an oxide layer underlain by an oxygen-stabilized α-phase layer. When cracks occur through the oxide scale, the uptake of hydrogen by the zirconium accelerates. "Today it is well understood that the fast embrittlement of claddings is mainly caused by breakaway oxidation… which leads to enhanced oxidation and hydrogen uptake and thus embrittlement." [144] If Zircaloy-4 cladding increases to temperatures of 950-1050ºC, then breakaway oxidation may initiate because of the formation of layered and cracked oxide scales that open up oxidation pathways. Stressed embrittled cladding can initiate cracking on the external surface of the cladding.[51] Fortunately, such temperatures are not expected during dry cask storage.

Over the last 20 years, studies have provided conflicting arguments about what combinations of conditions are necessary to initiate the formation of radial hydrides. What is known is that hoop stress, cladding temperature, and irradiation effects are important. The current approach is to limit the hoop stress in the cladding to preclude the formation of radially oriented hydrides. In general, a temperature limit of 400ºC is specified for normal conditions of storage, short-term fuel loading, and storage operations (which include drying, backfilling with inert gas, and transfer to the storage pad) to limit cladding hoop stresses and limit the amount of soluble hydrogen available to form radial hydrides. The NRC interim staff guidance on cladding (ISG-11) recommends that the change in cladding temperatures during episodes of thermal cycling (i.e., drying, backfilling and cask-transfer procedures) be restricted to "65ºC and minimizing the number of [drying] cycles to less than 10" to minimize the formation of radial hydrides.[145,150]

For the case of low-burnup (about 30-35 GWd/MTU) Surry PWR fuel rods kept in dry storage for 15 years in a Castor V/21 cask, there was no evidence of new hydrogen absorption, only "very few and very small radial hydrides present," [146] and no visible evidence of significant reorientation from the dominant long stringer hydrides, which were oriented circumferentially, to radial hydrides. Since "the stress required for reorientation appears to increase with decreasing temperature, and the stress in the fuel rods decreases with decreasing temperature [during dry storage]… any additional hydride formation during further cooling should also be circumferential,"[146] which should not lead to degradation problems.

As reported in the previous subsection on creep rupture, Argonne National Laboratory investigated segments of defueled high-burnup (about 67 GWd/MTU) Zircaloy-4 cladding from the H.B. Robinson PWR plant and found "significant waterside corrosion and hydrogen uptake…. One Robinson sample, which ruptured in the weld region at 205ºC during cooling from 400ºC under stress (190 MPa or 27.6 ksi), precipitated all visible hydrides in the radial direction."[137] These problematic large radial hydrides were seen all over the C15 Robinson rod segment as shown in Figure 35. A leakage failure was found in the cladding at its top end-cap fitting where it was welded. These segments all had circumferential hydrides before the creep tests at the increased temperatures and subsequent cooling, which suggests that more research would be helpful to better understand the conditions in which radial hydride form, and from which new limiting conditions could be employed to prevent radial hydride formation and its effects during dry storage.

Recently, Korean researchers re-analyzed previous experimental work and carried out new cladding tests to explore this phenomenon. They concluded, "prior plastic deformation" occurring as a result of creep during vacuum drying "promotes the nucleation of hydrides… which thus far has not been appreciated."[128] In particular, the size of the reoriented hydrides becomes larger with increasing peak temperature. The higher temperatures allow more creep deformation and experiments indicate that "the higher the prior creep strain becomes, the larger the radial hydride" that is precipitated. Figure 35 shows the result after a prior creep strain of about 3.5 percent; another test with creep strain of less than 1 percent precipitated only small fine radial hydrides. These same results were observed for different Zircaloy claddings and for both irradiated and unirradiated cladding. In Figure 36, the creep deformation at 300ºC was smaller than that at 400ºC, which tended to suppress precipitation of the radial hydrides. There also is a stress effect: higher cladding hoop stresses result in larger radial hydrides (again, higher stress means more creep strain at the elevated temperature).[128]

If dried rods are returned to the reactor pool and if they are still relatively hot, when they are lowered into the water, the rods will experience a rapid quench cool-down, which may induce high stresses in the cladding. Fuel rods have failed during quenching and handling of the rods as a result of hydrogen embrittlement. "Cladding failure mechanisms and probabilities will depend on cladding embrittlement."[147] The hydride embrittlement phenomenon increases with burnup. High-burnup effects in Zircaloy-4 cladding, and especially new claddings, are an important area of continuing research. EPRI has coordinated research with high-burnup fuels and new claddings for a number of years, but the results are largely proprietary.[88]

Conducting more research is important to reduce the uncertainty associated with the variety of conditions described above and their influence on cladding embrittlement.

Delayed Hydride Cracking. DHC is a phenomenon that starts with the cladding material becoming more brittle, the continuing formation of a hydride zone, and then existing stresses that can lead to progressive cracking of incipient cracks. The radial hydrides that precipitate are oriented parallel to (see Figure 35), and are often found near or surrounding, the crack tips of preexisting incipient cracks.[148, 149, 150] One finite element modeling study found “a compressive stress exists at the inner side of the cladding, while a tensile stress is found at the outer side of cladding, thus resulting in crack propagation from the outer side to the inner side of the cladding.”[59] DHC in zirconium alloy occurs as a consequence of three processes: nucleation of radial hydrides, growth of these hydrides to a critical length, and in situ stresses leading to creep and crack growth.

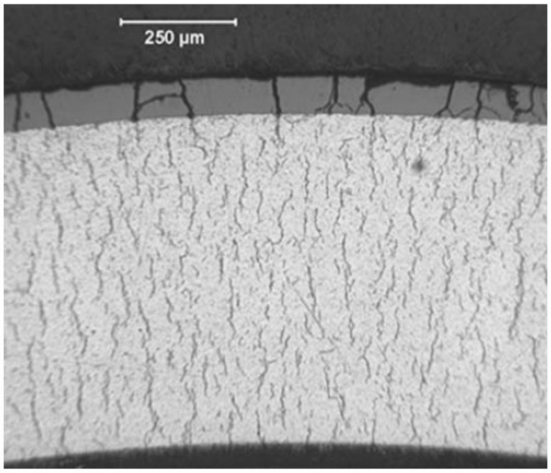


Figure 35. Large Radial Hydrides in High-Burnup Fuel-Rod Cladding After Cooling[137]

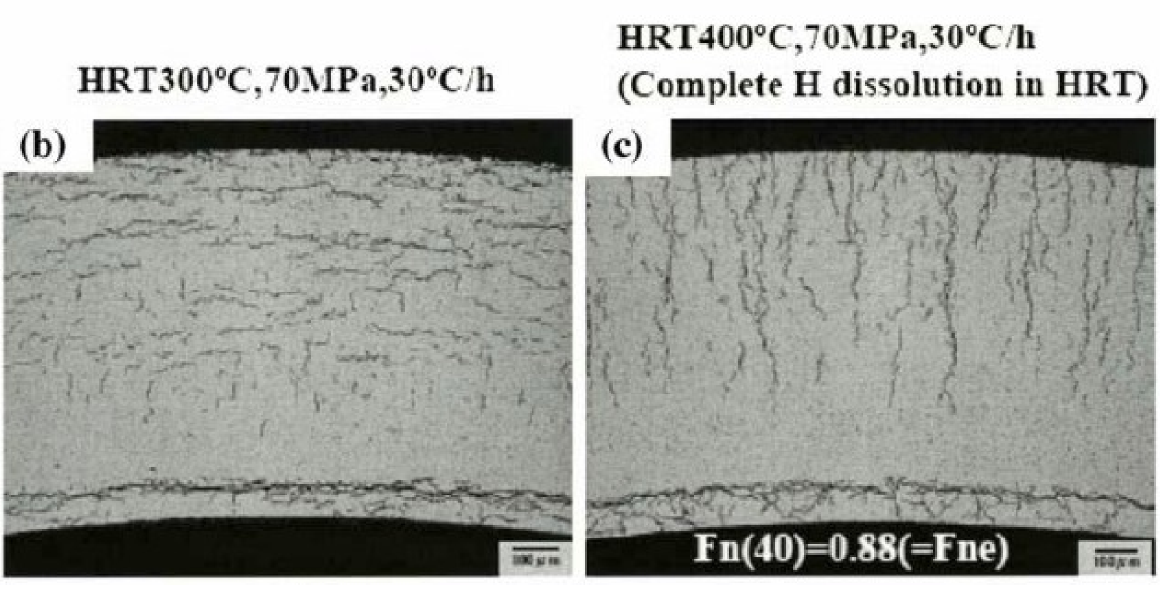


Figure 36. Distribution of Hydrides of Irradiated Zircaloy-2 Cladding After Heating and Cooling[137]

Previously, DHC was considered not to be a significant degradation mechanism during extended storage because cladding stresses near a defect were expected to be less than a threshold stress intensity.[143] However, the leader of the Zirconium Team with Korea's Nuclear Material Research Division pointed out that "little attention has been paid to delayed hydride cracking (DHC) of used fuel rods [at low temperatures]. The reason is based on the invalid hypothesis that DHC will not occur in the spent fuel rods due to limited stresses and slow diffusion of hydrogen at low temperatures below 200ºC."[128] The crack growth rate is governed by the rate of the slowest of the three DHC contributing processes stated above, which he claims is the nucleation process at low temperatures. He cites several instances of used-fuel-rod failures with leaking that can be attributed to: (1) the presence of existing residual stresses (e.g., a welded area of cladding) and incipient cracks, and (2) "stress-induced hydride phase transformation from γ to δ at a crack tip, creating a difference in hydrogen concentration… which is a driving force for DHC."[128,151] Because this hydride phase transformation and DHC are not observed until used fuel rods cool to below 180ºC (and it is predicted that fuel will take at least 30 years to cool to this temperature), they conclude that "we need considerably long cooling times before reaching a conclusion regarding whether the spent fuel rods will fail by DHC."[152] Extended discussion on this topic appeared recently in the *J. of Nuclear Materials* and focused on differences in theoretical thermodynamic interpretation, but the material issues of concern remain unresolved.[153]

The fine pin hole or hairline through-wall cracks that form from DHC may release internal rod He pressure, reducing the hoop stress acting on the crack and arresting further DHC cracking (the gross fuel-rod integrity may be maintained). However, if helium gas can get out, then air with oxygen may enter, causing UO_2 oxidation of the fuel, fuel expansion, and a gross cladding rupture. The rupture occurs as a result of long axial DNC crack growth, which can be expected to "occur only after cooling to below 180ºC in dry storage."[128]

NRC's current position, as expressed in ISG-11, is that "even if hydride reorientation occurred during storage, the network of reoriented hydrides is not expected to be extensive enough in low-burnup fuel to cause fuel-rod failures."[154] This recent Korean research may indicate that further experimental research is needed to confirm whether hydrogen embrittlement (as discussed above) or DNC leads to fuel-rod failures during dry storage, handling or transport. Careful characterization of used fuel during the storage life cycle can better indicate what threat DNC may pose to safe dry storage, handling, and transport. The current temperature constraint of 400ºC to limit the maximum cladding hoop stress may need to be supplemented in order not to exceed a maximum allowable creep strain for the cladding. The trend toward higher-burnup fuels results in more initial cladding damage and higher potential temperatures and stresses. This and the trend toward longer dry-storage times suggest a need to better understand and validate cladding behavior over time.

d. **Low-temperature embrittlement**. At higher temperatures, metals act in a more ductile manner than at lower temperatures. If the failure stresses are reached at lower temperatures, failure can be sudden (brittle), especially with sudden impact. With storage of used fuel for extended periods, cladding temperatures may drop sufficiently to make low-temperature embrittlement a significant phenomenon. Research is needed to further understand the significance of this phenomenon.

e. **Radiation effects.** In-reactor irradiation typically results in radiation hardening caused by an increase in strength and a decrease in both ductility and fracture toughness. Out of the reactor, β and γ radiation predominates, which does not cause much damage to Zircaloy cladding or significantly change its mechanical properties. The "radiation doses received during dry storage will be approximately four to seven orders of magnitude lower than received during reactor irradiation. Therefore, no new radiation damage is expected to occur in dry storage."[48] But this damage is believed to be cumulative, so what happens over another several decades is unknown. Whether low levels of ongoing radiation could significantly participate in cladding degradation is a problem that needs to be explored with observation and testing, given the long storage time in an active radiation environment.

f. **Cladding thermal annealing.** Under the elevated temperatures of the fuel-drying process, the metal cladding could anneal (soften). This could reverse some of the Zircaloy hardening caused by irradiation damage. However, inspection and testing of low-burnup (about 30-35 GWd/MTU) Surry reactor PWR fuel rods after 15 years of storage showed no significant annealing effect (initial rod temperature of about 350ºC with temperatures reaching 415ºC during less than 72 hours of performance testing). Further, the literature suggests that cladding held for thousands of hours at 323ºC experienced no significant annealing effect. Such an effect is expected at temperatures above 400ºC.[155]

g. **Zinc vapor chemical reaction**. Some casks have been painted on the inside with a common corrosion-resistant primer that protects steel galvanically, named Carbozinc 11 or have a different coating that includes zinc. From Table 1, the storage-only systems with zinc include TN-24, TN-40, VSC-24, and MC-10. At modest temperatures, a zinc vapor has been discovered to form within the cask system that can react with Zircaloy cladding and perhaps other storage system materials. The Zircaloy reaction product "ductility and strength properties are not as favorable as those of the… [original] cladding."[48] Further research is required to determine how significant a problem zinc vapor may be for these casks. After 20 years, temperatures have dropped to low values, so it is unlikely that new zinc vapors will form to cause problems.

h. **Pitting corrosion.** Modern welds on cladding have excellent pitting and crevice-corrosion resistance. Assuming that the inert gas atmosphere is maintained, pitting or other cladding chemical corrosion should be insignificant during long term-storage. If the cladding is breached in an oxidizing environment, experiments indicate significant degradation (corrosion) of the fuel may occur assuming a loss of helium canister cover gas. Fuel-degradation products will likely generate hoop stresses in the cladding sufficient to result in axial splitting of the cladding, following the mechanism discussed above in the subsection on Zircaloy cladding tangential creep rupture.[156]

i. **Biocorrosion.** Biofilm formation, biofouling and microbial influenced corrosion all have been observed in nuclear power plant water cooling systems and reactor storage pools. Studies have shown that “many exposed microbe species survived and even flourished in these high-radiation fields, which suggests that these radiation-resistant microbes could produce biofilms on waste storage containers or on used-fuel cladding and influence the corrosion of these materials.”[157] These microbes probably were introduced into the CSNF storage pools by windblown soil. The observed formation of biofilms, implicated in enhanced corrosion of metallic materials, demonstrates that microbial-influenced corrosion is possible on waste-storage containers or on used-fuel cladding. Whether these microbes have been eliminated in the drying process is not known, but if they were in the water, little remains for active degradation. Research is needed to better understand these mechanisms and their response to radiation fields and the inert gas environments of dry storage.

j. **Combinations of mechanisms.** Research generally focuses on one or the other of the fuel-degradation mechanisms noted above and does not take account of all or combinations of these above mechanisms acting simultaneously or in a coupled manner. There are several ways in which one type of degradation can initiate or accelerate other types of degradation. For example, in-reactor fretting wear or hydride breaches could lead to oxidation of the cladding (Zircaloy reacts with water to produce an oxide plus hydrogen). The hydrogen can be released through the cladding hole and react with Zircaloy at other locations on the fuel rod, causing hydride precipitation during the drying process. This could lead to embrittlement and other failure features and possibly a gross failure. Another example is the effect that irradiation of Zircaloy cladding material and other structural components have on hydrogen solubility in the Zircaloy materials. Low-temperature brittleness is likely to influence delayed hydride cracking and stress corrosion cracking. Some combinations of mechanisms are possible in a helium environment and others may be activated if the helium is lost from the canister, which reinforces the need to ensure that a helium cover gas is maintained during long-term storage.

Summary – degradation behavior of used fuel cladding: The key cladding degradation mechanisms noted above are known. What has not been sufficiently studied is the effect of these degradation mechanisms under the conditions of very-long-term storage. Although analytical models can predict consequences of individual mechanisms fairly well, coupled models may be needed to better predict the effects of multiple mechanisms. To validate these models, it is important to inspect representative low- and especially high-burnup used fuel after 30 to 40 years of dry storage.. Creep rupture is a known degradation mechanism that should be researched for better understanding of how much creep occurs in actual stored cladding over time, especially for high-burnup fuels, and to validate creep modeling. Creep that occurs at lower temperatures may be the result of a different mechanism from that of high-temperature creep. Stress corrosion cracking is another mechanism that is believed to be active but needs to be better understood. Hydrogen-related degradation mechanisms and effects are a recent renewed area of research with the potential of having significant effects on cladding degradation over time. The hydriding phenomenon is important from the perspective of embrittlement of the cladding which would affect its integrity during transportation to either a reprocessing plant or a repository. One of the key findings is the importance of the helium cover gas in the canisters in limiting and potentially avoiding certain key degradation mechanisms. The extent of cladding degradation over long periods is an important aspect of subsequent used-fuel handling systems and transportation concerns.

4.4.3. Behavior of Dry-Storage System Components

This subsection discusses the possible degradation behavior of dry-storage system components other than fuel pellets and cladding. These components, their function, and material description were discussed in subsection 4.1.3. Many of these potential degradation mechanisms are understood and well documented as a result of many years of research and observation of metals and concrete. Some of the material degradation mechanisms may require further study to predict the outcome under specific conditions present during extended dry storage. In any case, the design-life goals and analyses used by designers of dry-storage systems that have received licenses to date will need to be revisited to obtain renewed licenses.

Unlike the used fuel and cladding, several dry-storage components can be accessed for inspection and maintenance. This means that many of the potential aging problems that may develop should be identified and corrected before they cause any significant safety or performance concerns.

4.4.3.1. Metal Canister and Structural Components

The metal canister and structural components have been described in subsection 4.1.3.1. Of particular interest are the welds and metal near the welds. The safety classifications and functions of these dry-storage system components have been referred to previously. The potential degradation of any of these components is important to understand and acknowledge so that the functional and safety consequences can be fully known. This subsection will survey a number of important potential damage conditions and mechanisms affecting metal canister and structural components.

a. **Atmospheric Chemistry Corrosion Environment.** A dry-storage nuclear waste cask or canister exposed to the atmosphere for decades to a few centuries will be exposed to a corrosion environment determined by virtually all chemical species in the atmosphere and the effects of gamma radiation. This exposure will be enhanced by the natural convection cooling of the canister where air, along with dust, will flow along the metal surface. The flow of air will continually bring dust into contact and deposition with the metal surface, and the dust will contain chlorides, nitrates, salts, insoluble inorganics, and organics. Gamma radiation will be present and will result in chemical reactions occurring in the local air and dust; the result will be the production of a spectrum of chemical species. The question that must be considered with respect to a long- and very-long-term dry-storage container in air is, what will be the chemical composition of dust on the metal surface as a function of time?

   Organics are ubiquitous in atmospheric dust and the literature abounds with observations of organics.[158] Nitrates and chlorides are also present in virtually all dusts.[159] As a result of nitrates and organics both being present in dust, a hypothesis is that nitrates, which can inhibit corrosion, will react with organics and disappear. This appears to be the case where atmospheric dust was heated and the depletion of nitrate was observed.[160] Hydrogen chloride, sulfuric acid, nitric acid and virtually all oxides of nitrogen occur in the atmosphere as a result of chemical reactions. These three acids will react with calcite and dolomite (magnesium, present in dust) and produce the corresponding chloride, nitrate and sulfate salts.

Coupled to the chemistry described above will be a gamma field at the metal surface of a dry-storage container. This gamma field should be of interest for susceptibility to stress corrosion cracking[161] because of gamma-field-induced chemistry changes. Observations from the used fuel test at Climax revealed that carbon-steel borehole liners that had contained used-fuel showed much more evidence of external corrosion near the top of the fueled section than those containing electric simulators.[162] In all likelihood, this corrosion resulted from the production of nitric acid and other oxidizing chemical species by the irradiation of moist air. The production of nitric acid in moist air outside a 21-PWR waste package has been quantified,[163] and the results of this particular work can be extended to estimate the production of other chemical species. Also of interest with respect to how the corrosion environment can be influenced by a gamma field is where the principal radiation damage of alkali halides, such as sodium chloride, results in the formation of bubbles of fluid halogen.[164]

There is no simple description of atmospheric chemistry and the components of dust. Therefore, there must be a concerted and focused effort to investigate how the atmosphere can determine or affect the corrosion environment on the metal surface of a long- and very-long-term dry-storage canister. The investigation must include the presence of gamma radiation.

A similar environment and many of the same concerns will apply to external concrete in dry-storage systems. Both the exterior of the concrete cask that houses the metal CSNF canister and the concrete foundation pad are exposed to humidity, airborne salts, and the effects of wind-blown dust. For all cases, it will be important to be able to predict the surface canister temperature over the period of dry storage and validate this modeling with actual cask system measurements.

b. **Pitting and crevice corrosion.** For a bare-fuel storage system, there is a moderate risk for a failure of metal cask sealability. There have been two reported cases of failure of secondary (outermost) gaskets due to corrosion.[165] One of the failures was due to pitting corrosion; the form of corrosion for the other failure was not determined. For a canister-based storage system there is an insignificant risk of canister wall failure due to pitting or crevice corrosion unless the duration of storage is very long. A metal canister's corrosion lifetime for pitting and crevice corrosion can be estimated as the time needed for the canister to decline to a temperature where corrosive aqueous environments are possible (i.e., below about 100ºC), plus the time for electrochemical corrosion to propagate through canister wall thickness. One set of natural exposure tests (Japan, 60ºC, 95 percent relative humidity, saturated NaCl steam mist) on type 304 stainless steel yielded a maximum pitting penetration rate of approximately 20 μm/year and a maximum crevice corrosion penetration rate of approximately 30 μm/year.[166] Assuming a thin-wall canister thickness of 13 mm, these rates of pitting and crevice corrosion would lead to wall penetration in 650 years and 430 years, respectively, after the temperature on surface of the canister has dropped to the point where condensation on the surface can occur.

Furthermore, corrosion of external or exposed components of storage systems is either detectable to inspectors, or may be in the future, as new surveillance technologies are developed. Therefore, from routine monitoring, potential problems can be discovered and mitigated before too much damage occurs. Indirect indicators of corrosion could also be seen and addressed such as the presence of liquid water or salt buildup. For areas where direct observation is not possible, a means of remote inspection, indirect testing, or possibly a plan for periodically removing canisters for a physical inspection is important, particularly in harsh environments where external corrosion is possible.

c. **Galvanic corrosion.** Dissimilar metals are sometimes in contact with each other in some dry-storage systems (see subsection 4.1.3.1). When dissimilar metals are connected electrically in the presence of an electrolyte (liquid water or medium to high humidity), a galvanic cell is established and electrochemical reactions can occur. All casks contain varying amounts of residual water. For example, if bolts are anodic to large component of a cask system, the bolts may corrode quickly and lose their ability to function as fasteners in the system. For instance, if a relatively active metal like zinc, in the presence of large areas of ferrous surfaces that are cathodic to zinc, there may be a potential problem. It is important to systematically consider the various metals in use in a given dry cask storage system to anticipate galvanic corrosion. Coatings, electroplating, weld overlays, and metal seals are all a source of dissimilar metals.

d. **Stress corrosion cracking (SCC)**. SCC in metals can occur from the combined effect of tensile stress and the presence of a water-soluble chloride salt. Bulk liquid water need not be present because an aqueous environment will always exist on metal surfaces exposed to the atmosphere due to sorption of moisture from the air. This moisture sorption phenomenon is sometimes called "physisorbed" water.[81] This type of corrosion also is referred to as chloride-induced stress corrosion cracking. SCC is considered corrosion with local slip at the crack tip and is often is found to initiate where pitting or crevice corrosion has occurred. Air cooling of stainless steel canister walls means that the wall is in direct contact with moist air that will contain salt, including chlorides. Austenitic stainless steels are susceptible to chloride stress corrosion cracking starting on the outside surfaces in certain humid marine air environments under a tensile stress. Residual tensile stresses in the storage canister are mostly derived from welding between the wall and lids and cold working of the metal. As the temperature of the canister decreases, condensation can occur resulting in the accumulation of corrosive media such as chlorides, particularly in marine or industrial environments.

A metal canister's corrosion lifetime for SCC corrosion can be estimated as the time needed for the canister temperature to decline to the point where condensation is possible, which can be above 100ºC, plus the time for SCC corrosion to initiate, plus the time needed for SCC corrosion to propagate through the canister wall thickness.

A series of natural exposure and accelerated corrosion tests of conventional stainless steel used fuel canisters were conducted recently by the Central Research Institute of Electric Power Industry in Japan recently.[166] Natural exposure tests were conducted at Miyakojima, an island, one of the most corrosive areas in Japan, and accelerated corrosion tests were conducted in an environment filled with NaCl steam mist at 60ºC" and "humidity: 95 percent RHS.[166] One set of experimental tests on types 304 and 316 stainless steel yielded SCC corrosion initiation times ranging from about 1.6 to 3 years under natural exposure conditions. Pitting or crevice corrosion was found to be a trigger to SCC because SCC initiation was found to start from the bottom of the corrosion area by pitting or crevice corrosion. Under natural exposure conditions, the SCC corrosion rate varied from about 0.04 to 0.6 mm/year over a range of residual tensile stresses, "which would take from about 25-375 years to penetrate through a thickness of 15 mm" (0.59 in.) after the temperature on surface of the canister has dropped to the point where condensation on the surface can occur and SCC has initiated."[166]

Other Japanese researchers found that at or below the critical relative humidity of 15 percent, no stress corrosion crack propagation in 304 stainless steel occurs (2 mm thick plates, 11 mm square) after 125 days and some applied stress.[167] This same group of researchers determined that highly corrosion-resistant stainless steels (S31260 and S31254) become immune to SCC when the temperature decreases to the critical value of 298 K (25ºC or 77ºF), below which the anodic process of atmospheric corrosion is negligible for a 60-year period of storage.[168] As such threshold conditions are better understood, more corrosion resistant materials could be employed for use in extended dry storage, or new dry-storage designs may be used to prevent threshold corrosion conditions.

e. **Irradiation damage to other container materials.** Generally, effects of irradiation are likely to be most severe for polymer materials and the least for metal. Evidence appears to link irradiation with hardening and less ductility of Zircaloy grid elements. The "radiation doses received during dry storage will be approximately four to seven orders of magnitude lower than received during reactor irradiation. Therefore, no new radiation damage [referring to cladding] is expected to occur in dry storage."[48] Whether low levels of ongoing radiation could significantly degrade various container materials is a problem that needs to be explored with observation and testing, given the long storage time in an active radiation environment. Subsection 4.4.3.2.i. provides a more detailed discussion of this consideration.

f. **Component specific degradation.** Some components are more susceptible to corrosion and degradation than others because of the high level of residual stresses, high stress loads, abrasion, dripping water, or similar special circumstances.

- Welded areas often will induce high residual stresses unless post-weld-stress-relief heat treatment has been performed. Besides stresses, the heating from welding can make the area especially susceptible to corrosion. Records of weld examinations and photos may be helpful.
- Bolts are subject to high stress. Under both thermal accident events and mechanical impact events, relatively brittle welds can fracture. Under accident thermal events, creep rupture failures also are possible.
- Metallic seals and gaskets under high loads can degrade through stress relaxation of the metal, plastic deformation, and creep.

Because of these possibilities and others, identifying all components that perform important functions in dry-storage systems and researching how the performance of these functions may degrade for extended periods of dry storage are important.

4.4.3.2. Nonmetal Components within a Canister

Important components reside inside canisters that are made from a variety of different materials. Any of these materials could degrade with time, especially early in dry storage when the radiation, heat, and corrosive atmosphere are highest.

a. **Shielding and neutron absorbing materials**. A number of materials whose function is to provide shielding or neutron absorption were previously described in subsection 4.1.3.1. Some of these materials are used as part of assembly designs that are subsequently placed in wet and dry storage, such as aluminum boron carbide cermet (BORAL), nonmetal matrix composites, METAMIC, Bortec, and Carborundum B4C plates. Shielding materials, such as borated polymers and Carborundum plates, must not creep, slump or move significantly from their original locations. A few of these materials have suffered degradation while in reactor storage pools; typically they were part of pool rack-storage systems.[169] This motivated a sustained EPRI-led program to understand and mitigate these degradation problems. BORAL materials and Carborundum plates experienced blistering and bulging due to gas accumulation, but the material performance was reviewed and found acceptable for use in dry-storage systems (degradation was linked to the pool environment). When dry-storage systems are opened and inspected, shielding and neutron-absorbing materials should be checked for any aging degradation to verify that no problems develop in extended storage.

b. **Component-specific degradation.** Other components may also experience aging. For example:

   - Radiation can cause polymerization (cross-linking) in elastomer seals and lead to embrittlement. Radiation and heat also can cause chemical breakdown, releasing decomposition products that may react with other components, for instance, corrosive fluorine gas. Replacement schedules for elastomeric seals and other components should be identified in the system Safety Analysis Report (SAR).
   - Ceramic cask components can be damaged through impact, sustained loads, aging, and other processes.

4.4.3.3. Concrete Structures and Components

Concrete is used in the overpack of the canisters and in the foundation pads for the stored canisters. The degradation of concrete is caused by many physical and chemical mechanisms, some of which act in concert. The age-related mechanisms that may degrade concrete and reinforcing steel during dry storage are discussed in this subsection. Aboveground concrete structures that are exposed to the environment are expected to degrade over time. All concrete used in a dry-storage system is reinforced, regardless of purpose, and usually 5,000 psi minimum compressive strength concrete is specified. Type II Portland cement typically is used because of its higher sulfate resistance and lower heat of hydration in comparison to general purpose Type I cement. Other degradation mechanisms may be experienced by concrete in contact with the ground, such as the foundation pads, because of groundwater and chemicals in the soil.

Potentially significant degradation mechanisms include the effects of elevated temperature on concrete, freezing and thawing, and the leaching of calcium hydroxide or other soluble constituents.[170] Reinforced-concrete structures for dry-storage systems can be designed with a lifetime of 100 years and longer.[171] Research is needed to determine the duration of dependable design lifetimes. Periodic inspections would be helpful to identify and correct potential problems.

a. **Elevated temperature**. Concrete that is exposed to high temperatures will lose moisture and eventually can induce thermomechanical destruction of both the internal cement structure and the bonds between cement and aggregate. Undesirable chemical reactions between the cement and aggregates also can occur. All of these types of degradation significantly decrease concrete strength, elastic stiffness, and toughness. For example, at 150ºC (300ºF), the mean concrete strength is about 80 percent of its normal value. Generally speaking, the threshold of degradation in the concrete is at a temperature range of 66ºC to 95ºC (150ºF to 180ºF). Because the high-level wastes and CSNF may reach a temperature in the range of 150ºC to 180ºC (302ºF to 356ºF), the concrete would need to handle such temperatures as well if natural cooling mechanisms fail. Exposure to elevated temperature could be a significant age-related degradation mechanism for the concrete enclosures. Creep of concrete at an elevated temperature under a sustained load could have an adverse effect on the concrete structure over time.[172] Fortunately, the loads on the concrete overpacks are limited to their own weight. Concrete cask overpack material will be the most susceptible to this kind of damage due to the close proximity to the heat-generating fuel. Expected interior-surface concrete temperatures could be estimated in thermal calculations to better evaluate this degradation mechanism. Furthermore, if air vents are plugged because of some natural or human-caused off-normal scenario, concrete temperatures could rise significantly above the normal operating range.

b. **Freeze-Thaw Cycles.** Concrete that is located in climates where temperatures plunge below freezing and are relatively wet, or where water is in the soil, are potentially susceptible to freeze-thaw damage. If liquid water pulled into concrete pores by capillary action freezes, then large expansive stresses can be induced in concrete pores (hollow cast-iron cannonballs can be fractured if filled with water that freezes). Freezing also can force adjacent liquid water into pores and cracks, and the water can subsequently freeze. Cycles of freeze-thaw are an effective mechanical mechanism of concrete degradation, especially near the surface, that results in cracking, scaling and spalling. If concrete degradation is severe, then rebar can be exposed, probably initiating accelerated metal corrosion. "The primary parameters that affect the occurrence of such degradation in concrete include the air content of concrete and the number, size, and distribution of the pores with the aggregate of the concrete…. Degradation due to freezing and thawing may be a significant issue for surface concrete storage casks stored on the surface when the heat load of the SNF has dissipated."[173] More-porous concrete and higher-performance (more strength and durability) concrete can be used to mitigate the freeze-thaw mechanism. For anticipating the severity and frequency of freeze-thaw cycles, the climate at a particular site will have to be evaluated. The earliest failure times will be for cold climates, such as that of Minnesota. "Analysis of NUHOMS dry-storage casks show they may fail after 80 years in cold climates if they are not maintained."[174] In warm climates, such as in coastal California, this degradation mechanism will not apply.

c. **Leaching of Calcium Hydroxide**.[175] Climates with significant rainfall provide a source of flowing water on concrete surfaces and in cracks or joints that can leach out calcium hydroxide (white hydrated lime) from the concrete. Removal of calcium hydroxide can significantly weaken concrete mechanically, and leaching over long periods increases the porosity and permeability of concrete, making it more susceptible to leaching and freeze-thaw damage. Damage from calcium hydroxide leaching can be suspected when white leachate deposits are observed on the surface of concrete. Concrete that is exposed to groundwater also may be susceptible to calcium hydroxide leaching, but this damage may not be discoverable unless soil is removed for inspection. Inspection of the suspected degraded surfaces should reveal the degree and extent of degradation.

d. **Chemical Attack**. Chemical attacks on concrete derive from hydrated carbon dioxide in the air and hydrated sulfates in the soil. Both can lead to local acid solutions in concrete that react with the high alkalinity calcium hydroxide (lime) of concrete ($pH > 12.5$) to convert the calcium hydroxide to calcium carbonate (a process called "carbonation"). Fully "carbonated" concrete results in concrete pH levels slightly above 8. The carbonation progression inward in concrete proceeds at a slow but fairly steady rate, on the order of 1-2 mm per year.[177] Whereas carbonation tends to decrease porosity and increase concrete density and strength, sulfate attacks usually increase the porosity and permeability of concrete, and can lead to decreased density and compressive strength. Sulfate reactions are accompanied by an expansive stress within the concrete, which can lead to spalling, cracking, and strength loss.[176] The presence of chlorides also lowers the pH of concrete and so can lead to corrosion of reinforcing steel. For concrete that is obscured from view by soil sulfate attacks may not be noticeable until damage is significant.

e. **Corrosion of Reinforcing Steel**. For the reinforcing steel bars in concrete to corrode, diffusion of both chloride ions and carbon dioxide into the concrete is required at particular concentration thresholds. The high alkalinity of concrete ($pH >12.5$) protects the embedded reinforcing steel from corrosion. However, when the local pH near the steel is decreased below the threshold level of about 11.5, then chemical corrosion initiates more easily. Initiation of chemical rebar corrosion also requires a chloride threshold concentration, which is on the order of 0.06 percent by weight of concrete for black steel.[177] The pH reduction occurs by intrusion of aggressive ions, primarily chlorides, in the presence of oxygen. The transport of chloride ions in intact concrete occurs by ionic diffusion according to Fick's Second Law of Diffusion. This means that the higher the chloride concentration in the seawater next to concrete, the faster the chloride ion diffusion will occur in the concrete. Accordingly, concrete surfaces that are continually exposed to salty marine or industrial chemical environments are most at risk, particularly if cracks are present. "Calcium chloride accelerates the corrosion more than sodium chloride…. In addition to the corrosive agents, the severity of corrosion is influenced by the quality of concrete (cement type, properties of aggregates, and moisture content), depth of concrete cover over steel, and the permeability of concrete."[178]

The key outcome of this degradation mechanism is the creation of steel corrosion products that cause high swelling pressures within the concrete, which often leads to concrete spalling and cracking. Once rebar corrosion initiates, significant concrete near-surface damage can proceed quickly (on the order of months or a few years). This minor spalling and cracking usually will not affect the structural performance of the reinforced concrete significantly until a critical level of rebar tensile strength is lost. The service life of concrete structures where this is the dominant degradation mechanism, and it often is, can be predicted by models based on the chloride ion diffusion. Because rebar is hidden from view by the concrete cover, any initiating degradation of concrete structures by this mechanism can be detected by making electrochemical measurements of the reinforcing steel that indicates corrosion, if this capability was designed in advance (none of the current dry-storage systems relying on concrete have this capability).

f. **Reaction of Aggregates with Alkalines**. The alkaline aggregate reaction is a common concrete-degradation mechanism identified by an irregular cracking pattern on the surface of concrete. The mechanism can be greatly minimized by a careful selection of aggregate that does not contain reactive materials. "Two types of reaction, alkali-silica and alkali-carbonate, have been identified. Moisture must be available for the chemical reactions to occur. Thus, concrete that is either consistently wet, or that experiences wet and dry periods, is susceptible"[179] Most reactive aggregates have been identified in the construction industry and should be avoided.

g. **Creep and Shrinkage**. A normal part of concrete behavior, as concrete dries out, is that it shrinks in volume, inducing tensile strains and, possibly, shrinkage cracks. "Most of the shrinkage (98 percent) typically occurs during the first few (e.g., five) years of service. The significance of a shrinkage crack as a potential contributor to degradation depends primarily on its size and environmental exposure conditions. A crack can allow aggressive agents access to the reinforcing steel, promoting the possibility of corrosion."[180] Shrinkage cracks can be avoided by following the recommended concrete-design practices.

h. **Irradiation**. "The degradation of concrete exposed to neutron and/or gamma radiation is manifested in many ways. Fast and slow neutrons usually cause aggregate growth, decomposition of water, and warming of concrete. Gamma radiation affects the cement paste portion of the concrete, producing heat and causing water migration. The degradation due to nuclear heating and water loss is more serious than degradation associated with direct radiation damage. This is because nuclear heating causes the free water within the concrete to evaporate, and both the neutron shielding and structural characteristics of the concrete become impaired. As a consequence, the concrete could experience a decrease in its strength (compressive, tensile, and bonding strengths) and stiffness (modulus of elasticity) from shrinkage and cracking if the thermal gradient is excessive. According to the American National Standard ANSI/ANS-6.4-1985, nuclear heating can be neglected if the incident energy fluxes are less than 1,010 MeV/cm^2-sec."[181] Under typical dry-storage conditions, the energy flux is negligible and decreases with time, making this aging mechanism insignificant.

Summary: degradation behavior of dry-storage system components. As is evidenced by deteriorating concrete bridges, atmospheric corrosion and degradation of concrete structures do occur. The additional influence of heat and radiation damage can compound environmental damage. Fortunately, the loads on the concrete overpacks are limited. In addition, the external surface of the concrete overpack and foundation pad, as well as a number of other dry-storage system components, cannot easily be inspected. Other components that are housed inside of welded containers also cannot be easily inspected. Aging of these component materials should be studied to better understand the possible extent and under what conditions they age. Monitoring the condition of concrete overpacks to identify damage before it becomes significant will be important. Once damage is found, a follow-up maintenance program will correct the damage and help minimize future degradation problems. There are already reported observations of visible metal and concrete degradation of dry-storage systems in areas near the sea. Having an inspection program at all sites for monitoring degradation on all metal and concrete exterior surfaces to better understand how degradation evolves in various climates is important. The dust and chemical species that comprise an atmospheric-corrosion environment can affect several different mechanisms.

4.5. Potential of Criticality During Dry Storage

The potential for criticality in dry-storage systems is almost non-existent because the used fuel is removed from the reactor since it can no longer sustain the fission reaction. If a dry-storage system is loaded entirely with used fuel, the likelihood of a criticality incident is remote even if water moderation is assumed.[182] However, dry-storage systems are designed with additional neutron absorbers in the cask or canister to prevent any possibility for criticality even if loaded with fresh fuel. Listed below are the measures taken for preventing criticality.

a. **Moderator Exclusion.** Even if the used fuel in a dry-storage system were rearranged to its most reactive configuration, it still would not be close to sufficiently reactive to allow criticality without the presence of a moderator (water). Water is excluded from the dry-storage system by drying the loaded storage system before it is closed and sealed, filling the system with inert gas under pressure, and draining water (e.g., rainfall) away from the outside of the system.

b. **Low Used-Fuel Reactivity.** In most cases, fuel is permanently removed from a commercial light water reactor because it no longer has sufficient reactivity to sustain a nuclear reaction.[183]

   The reactivity of used fuel changes with time. In particular, the reactivity of used fuel decreases significantly in the first 100 years after its permanent removal from the reactor, principally because of decay of Pu-241.[184]

c. **Neutron Absorbers.** Dry-storage systems have criticality control materials in their baskets, often in the form of aluminum panels that contain boron. As long as the panels

> are not significantly degraded, criticality is not possible, even if the used fuel is completely flooded with water. Significant degradation of the panels is not possible if the inert gas (helium) atmosphere inside the container is maintained. Consumption of boron by neutrons produced by spontaneous fission or by (α,n) reactions is insignificant.
>
> Criticality control material will not degrade in an inert environment. Penetration of the dry-storage system would allow helium to escape and air to enter, which could cause corrosion when the relative humidity of the air in the system reaches approximately 70 percent.[185] Decay heat from the used fuel would prevent the relative humidity from reaching that level for many decades.

Summary – criticality during dry storage: Criticality is highly unlikely in dry-storage cask systems. This remains true even if the casks are fully flooded with water or if the boron neutron absorbers are degraded or removed, due to the low reactivity level of the used fuel and the configuration of the used fuel in the canister system. This report does not evaluate the potential of criticality during transportation.

5. Evaluation of Technical Bases for Storage and Handling of Used Fuel

The comprehensive evaluation of technical concerns associated with extended dry storage is a difficult exercise. The analysis must consider initial conditions, boundary conditions, transient conditions, initiating events, all active and significant physical and degradation processes, the uncertainties involved, and focus on the predicted state of the fuel, cladding, and dry-storage components over time. A proper evaluation, in a rigorous manner, to determine dry-storage safety at a particular point in time requires detailed knowledge of the concurrent condition of the fuel, cladding, and dry-storage components. This is the theoretical approach.

With respect to an engineering problem that is inherently uncertain, there are two fundamentally different approaches that may be employed to address the uncertainty: 1) the empirical/extrapolation/observational path and 2) the theoretical/modeling/validation path. An empirical approach is a traditional engineering method, especially in civil engineering. Observations are made and collected, that are then applied to a subject area such as concrete deterioration. In searching to understand patterns in the observations, empirical relationships are developed and may be extrapolated. Subsequent observations alert the engineer for the need to adjust the empirical relationships or initiate corrective action. A clear theoretical understanding of the phenomena involved is not necessary, though it is helpful. On the other hand, the theoretical approach requires sufficient knowledge of significant physical processes, appropriate conceptual models, and boundary condition values in order for an engineer to be able to quantitatively predict what will happen. Scientists and engineers are sometimes more confident in their predictions than the subsequent results would indicate, and for this reason validation exercises are necessary. Decision-makers may elect to use a combination of these two paths. The choice of approach will determine what data and informational needs are required, and what kind of observation activities and research are important to support and fund.

The most comprehensive methodology that can be applied for quantitatively understanding the relative importance of various aspects of dry-storage safety would be to carry out a full probabilistic risk assessment (PRA). A systematic PRA would assess the potential risks to the public from each possible event scenario involving the handling, storage, and transport of CSNF for each type of dry-storage cask at their site-specific locations. These individual PRAs would then be analyzed and combined to identify the dominant contributors to risk. This knowledge would provide realistic risk insights to all stake holders and allow decision-makers to focus on effective ways to reduce uncertainty and risk. A pilot PRA, which did not consider aging effects, transportation, uncertainty analyses, and several other relevant concerns, was recently carried out on a single dry-storage system at a specific site to illustrate applying PRA methodology to dry storage.[84] NRC staff developed a comprehensive list of relevant initiating events (with corresponding frequency of occurrence) and evaluated the risk (over 20 years) associated with each initiating event, where the risk was presented in terms of the probability of a latent cancer fatality. This is not directly applicable to evaluating whether NRC regulations are met. In 2002, EPRI performed a study to evaluate various methodologies applicable to carrying out a dry cask storage PRA.[186] Quantitative PRA work is beyond the scope of this study. The approach used here is to review the literature and carry out a qualitative assessment to identify key risk factors that are most likely to affect dry cask storage, handling, and transportation over extended dry storage.

Establishing long- and very-long term technical bases for safe handling, dry storage and subsequent transportation of CSNF requires a focus on the degradation processes or aging of important-to-safety, dry-storage system elements, namely, fuel pellets, the cladding, assembly structural elements, and the canister system or cask over time. At some point in the future fuel may have to be repackaged. The evaluations are qualitative and based on the best technical knowledge reviewed and consulted as part of this study.

5.1. Dry-Storage Technical Basis for Used Fuel-Pellets with Aging

The fuel pellets, after irradiation in a reactor, come out in a damaged and fractured state as described in subsection 4.1.1. However, fuel pellet material has not typically been oxidized into a powder and the intact fuel-pellet pieces still retain radioisotopes and fission gases. Because most cladding is assumed to be intact, general containment of the radioactive particles and gases also is assumed. During wet storage, it is observed that no significant damage occurs to cladding and so no further significant degradation of used-fuel pellets or pieces can occur (because temperatures are stable and the helium gas environment persists preventing oxidation; see subsection 4.1.2.2). During the drying process (subsection 4.1.2.3), temperature cycling occurs that can further damage and fracture the fuel pellets.

The key controlling condition over extended dry storage is whether the cladding remains intact, which provides the containment function for inert helium gas and prevents further oxidation of the fuel matrix. Of the four long-term degradation mechanisms that apply to used fuel-pellet material (discussed in subsection 4.4.1), all are unlikely to cause significant fuel degradation during long-term dry storage for many years in a weak to non-oxidizing environment. Under normal dry-storage operation, the used fuel-pellet material then cannot significantly oxidize until and unless both a cladding and canister breach occur. Subsections 4.4.2 (cladding), 4.4.3.1 (canister), 5.2 and 5.3.1 explain that each triggering scenario should be unlikely to cause a gross cladding breach unless a serious off-normal or accident event occurs. If this happens, then the situation should be able to be cleaned up and contained relatively easily, before intact used fuel-pellet material would have time to oxidize and cause further cladding damage (because the oxidation rate is very slow at ambient temperatures).

Two other cases should be considered. The first case is one in which used fuel has oxidized as a result of failed cladding and this condition is known before the fuel is placed into dry storage. For the case of gross cladding failure, this could pose a large hazard, but requirements would have led to the failed fuel being safely placed in damaged-fuel cans before being placed into dry storage. It should be noted that fuel-pellet material that has been canned does not mean the fuel is separately confined. Often one end of the can is an open screen. Over the period of extended dry storage, the corrosion analysis of the “can” would be similar to that carried out for the cladding (except for any dissimilarity in metal or temperature conditions) and is discussed in the next subsection.

The second case would be one in which used fuel has partially oxidized in failed cladding and this condition is undetected. Obviously it could not be a case of gross cladding failure. It is possible that this situation could occur, but the cladding flaw would very likely be tiny; otherwise, it would have been noticed during wet storage. The likelihood that the cladding could subsequently experience a major rupture during storage is unlikely and is addressed in the next subsection.

However, the state of used fuel alone cannot undermine the technical safety basis for long-term dry storage or handling. Additional research is needed for understanding how high-burnup fuels and other types of fuels degrade with time in the storage systems.

5.2. Dry-Storage Technical Basis for Fuel Cladding Integrity with Aging

The Zircaloy cladding that houses used nuclear fuel and helium comes out of the reactor generally intact but significantly altered from its original manufactured condition, as described in subsection 4.1.2. Fuel cladding, upon its arrival for dry storage, typically has incipient cracking, a deformed shape, is under a three-dimensional state of residual stresses, may contain radial-hydride precipitation (that may cause embrittlement), and has experienced various degrees of internal and external corrosion and fretting wear. In wet storage, there is fuel with small cladding pinholes and defects, but it is not common to find new pinholes or cracks that occur out of the reactor. Whereas damaged fuel is often inspected and analyzed, there is limited inspection and characterization of undamaged fuel-rods. As a result, the exact condition of used fuel placed into dry storage is not entirely known, and some cladding could have partial cracks or partial fretting wear. However, the likelihood of serious structural weakness is low; otherwise, more cladding failures would occur during handling and preparation for dry storage.

Relatively little experience and research is available to understand the effects of high-burnup fuel on the state of cladding in the post reactor environment and on subsequent cladding behavior during dry storage. It is understood that, in general, the higher the burnup, the more cladding damage results.

During wet storage, it is generally assumed that no significant damage occurs to the cladding as discussed in subsection 4.1.2.2. During the drying process (subsection 4.1.2.3), temperature cycling occurs that can load and change the cladding properties in significant ways, particularly through the formation of radial hydrides. In addition, drying can create incipient cracks in the cladding. After the drying process is complete, it is likely that the cladding is in a more damaged state than when it was placed into wet storage. However, NRC regulation guidance limits temperature extremes and other adverse conditions to minimize this additional damage.

Subsection 4.4.2 discusses the likelihood that any or some combination of mechanisms could degrade fuel cladding and possibly lead to leaks or a cladding breach during extended dry storage or handling. Given typical storage conditions during 20 years of dry storage, the following mechanisms are not expected to cause further significant cladding damage: creep rupture, stress corrosion cracking, radiation effects, cladding thermal annealing, zinc vapor chemical reaction, pitting corrosion, and biocorrosion. That only leaves one problematic mechanism related to hydride reorientation, embrittlement and delayed hydride cracking. New research indicates that this mechanism is active at low temperatures and may weaken Zircaloy-fuel cladding, leading to hairline cracks, especially in areas of higher in situ stresses where cladding has been welded. The problem looks to be more severe for high-burnup fuels. Actually, the zinc vapor chemical reaction mechanism is a relatively new phenomenon that is not well understood. However, this only affects storage systems painted on the inside with Carbozinc 11.

A number of the plausible accident conditions for handling and storage could cause significant mechanical loading shocks to the cladding. For instance, a drop or collision during handling, an earthquake, a vertical storage system tip over, or a terrorist bomb could cause one or more stored rods to rupture.

Given the robust nature of modern storage cask systems, it is unlikely that the fuel canister will be breached, even in a severe accident or natural event, and so the retained helium gas will prevent fuel oxidation and associated cladding unzipping. Consequently, the used fuel of a failed fuel rod or canned fuel (open on one end) should remain largely intact over long- and very long-term storage. On the other hand, the failed cladding may go undetected and, upon opening the canister, a potential exists of releasing oxidized fuel and radioactive fission gases to the environment, which would necessitate being ready with measures to control and contain contamination.. Alternatively, when used-fuel assembles are dropped during handling or vertical cask systems tip over in an off-normal scenario, the induced stresses may cause some fuel rods to fracture due to their current brittle and damaged state. If the latter situation consisted of a gross rupture and fractured fuel-pellets spill out, it is unlikely the fuel would be in a problematic (fine powder) oxidized state, so the particulate problem could be cleaned up and recanistered with relative ease (albeit with special equipment). Any radioactive gasses that are released will be lost unless still contained within an intact canister.

An important aspect of extended storage is the need to ultimately handle the used fuel after storage, and transportation, and to be able to satisfy the transportation requirements. Prior to opening the canister, the condition of the used fuel will be difficult to ascertain. Methods need to be developed to know whether the fuel is degraded to a point where opening the canister or waste cask will release radioactive materials. For plants that no longer have a pool (decommissioned sites) that might be needed for repackaging, or for preparation of transportation, dry cask handling techniques will need to be developed that will allow shipment of fuel in storage-only cask systems. Because hydriding is considered a potential major concern for extended dry storage, research is needed to explore and better understand the potential problems associated with hydrogen-related mechanisms and effects of cladding degradation, especially of hydrogen embrittlement and delayed hydride cracking.

In summary, given the damaged and unknown state of cladding at any point of time during dry storage, there is a possibility that accidents or handling could fail weakened fuel-rods. Furthermore, the uncertain hydrogen embrittlement and delayed hydride cracking mechanism could result in additional failed cladding. The likelihood of the latter mechanism increases with time. Accordingly, cladding failure during dry storage and handling could present significant hazardous conditions to workers and especially if the used fuel needs to be handled. While analytical methods can be used to predict future conditions of the fuel and cladding, it is recommended that a validation inspection program of both low and high-burnup fuels be instituted after 15 and 30 years of storage similar to that performed for Surry fuel by NRC and EPRI. This would provide additional confidence of our understanding of degradation phenomenon for extended periods of dry storage.

For the case of failed cladding *and* a breached canister, there is the possibility that corrosion mechanisms within the canister could be activated. The new cladding degradation mechanisms that could be significant include pitting corrosion, stress corrosion cracking, induced cladding stress leading to creep failure, the potential of slow fuel oxidation that may result in unzipping, and biocorrosion. Analysis of the possibility of a breached canister due to degradation is explored in the next subsection. However, a breached canister should be able to be detected based on the release of radioactive contaminants.

Given the initial-state condition of cladding when it is placed into dry storage (after the drying process), there may be fuel rods that are close to failure (defined as having pinhole and hairline cracks; undamaged fuel). A fuel-rod breach could potentially occur by two processes: impact forces causing a mechanical rupture, or thermal creep and rupture of embrittled cladding. The breached rod would depressurize and release radioactive gas, and possibly fines material, into the interior of the bolted or welded container. For this reason, a PRA approach would be used to calculate the percentage of fuel rods within a dry-storage cask that are expected to fail given an initiating event, instead of calculating the probability of failure for some “typical” fuel rod.

In summary, normal events are not expected to result in conditions that could breach the Zircaloy fuel-rods. However, there are a number of off-normal and accident scenarios (mechanical impact and fire-related thermal-mechanical loading) and natural phenomenon scenarios (seismic motions, cask tip-over) that, when coupled with hydride embrittled cladding and higher burnup cladding corrosion, could result in cracked or leaking fuel-rods.

Furthermore, under most of these same scenarios, it is not likely that the metal canister will be breached (next subsection). This is important because even if fuel rods are slightly breached, the canister helium atmosphere will be maintained and prevent fuel-pellet oxidation, unzipping, and further fuel-rod damage. Given a long enough time and ineffective or no monitoring, it is conceivable that both cladding and canister containment could be breached, creating a significant public safety problem. If the initial breaches are minor, then it may take months or years before sufficient pressurized helium is replaced with air to initiate significant internal corrosion.

5.3. Dry-Storage Technical Basis for Dry-Storage Systems with Aging

At the initiation of dry storage, all system components are essentially brand new and in pristine functional and physical shape. This includes the assembly grid system, the neutron absorbing and the shielding materials. One exception is the in situ metal damage and residual stresses that have been caused by welding. These areas are more susceptible to certain mechanisms of degradation, as discussed later. Because the canister has been dried out and inerted with helium, no internal corrosion is expected during dry storage, assuming the helium is contained. The exterior environment of dry cask system components is a different story, however, because these external metal and concrete surfaces are exposed to oxygen in the air, humidity and salts.

5.3.1. Metal Canister and Structural Components

The metal canister is the ultimate barrier to release of radioactive materials from used-fuel storage systems. Even if the fuel is degraded, the integrity of the canister will protect workers and the public in storage and in transportation. A complete analysis of the state of the dry-storage system requires that each critical component be analyzed separately as to how the component may age or deteriorate with time. For instance, the evaluation and analysis of a given component should include information such as: the exact material of composition; any specifications and standards used in fabrication of the component; the vendor; mechanical, thermal, and other relevant properties; surface finish and coatings; stress and strain history (including temperature excursions, pressure loadings, residual stresses and annealing, welding history); characterizations over time; contact with dissimilar materials; and exposure to atmospheric and climatic conditions. Detailed calculations of various behaviors will need to be carried out by specialists to properly discover a quantitative answer to degradation and future performance safety.

After the degradation, performance data, and current state of a given component are known, an evaluation as to the various important-to-safety judgments can be made. This includes consideration of thermal performance, radiological shielding function, confinement, subcriticality, structural integrity, and retrievability. To actually evaluate specific suitability and expected performance of these components to perform their required functional and safety objectives, a systematic and time-consuming application of various scenarios, conditions, and components is required. This work is too specialized and detailed to analyze fully for specific storage systems at specific sites under specific test scenarios as part of this technical background report.

Instead, the approach chosen is to consider key components and combinations of scenarios and conditions that either can lead to a performance “failure” or the overall uncertainty involved is high, meaning expert confidence is low that the component will perform its function or perform safely under the combined set of scenarios and conditions considered. The most obvious key component is the thin-walled metal canister, which houses the used-fuel assemblies and the inert helium atmosphere that limits internal corrosion and other oxidation processes and facilitates heat transfer.

Given a “normal” operational scenario of a dry-storage system sited in a humid area of the U.S. near the ocean (conducive to a halide corrosive environment), it seems likely that one or more degradation conditions and mechanisms described in subsection 4.4.3.1 could eventually cause significant problems, and possibly a minor breach (hairline crack) of the canister. Of the two general types of storage systems, the thin-walled, concrete-shielded storage systems with welded lids seems the most likely to be breached by corrosion mechanisms. In particular, austenitic stainless steel may be vulnerable to stress corrosion cracking (SCC) under certain circumstances. It is important to prevent penetration through the wall by SCC allowing for the loss of the helium cover gas that would initiate additional degradation mechanisms of the fuel cladding and used fuel-pellets.

To better understand SCC deterioration mechanisms, research would be helpful to learn more about the threshold value of chloride to initiate SCC, estimate the chloride accumulation transported by cooling air on the metal canister surface, and clarify other aspects of atmospheric corrosion. With this information, SCC analyses needs to be performed to determine the lifetime of the canister material against SCC penetration, given site specific conditions and degradation mechanisms. It can be reasonably assumed that the corrosion or deterioration problems may be able to be identified and fixed by replacing the damaged components or carrying out an entire repackaging of the CSNF. Research is needed to improve the ability to detect canister deterioration before failure occurs to effect corrective actions. At present, no inspections are carried out for canister-based systems due to lack of a current need and the difficulty of performing physical inspections. Since there are no adequate inspection and maintenance procedures in place to systematically identify and correct the above potential aging problems with dry-storage systems, it is important to plan for, establish and execute a regular inspection and maintenance program.

In summary, normal, off-normal, accident thermal loading scenarios (engulfing fire and blocked vents), and natural phenomenon scenarios (direct hit by meteorite excluded) are not expected to result in conditions that could breach the metal canister. However, certain mechanical impact scenarios (high drop) and corrosion mechanisms (given enough time in a facilitating corrosive environment) may breach the container. A canister integrity failure by mechanical impact would most likely be a handling accident after extended dry storage when the metal canister had experienced corrosion or stress corrosion cracking. Failure by atmospheric or galvanic corrosion would also be after long-term storage. The initial container breach may be small enough so that months and years may be required before sufficient pressurized helium is replaced with air to initiate significant internal corrosion.

5.3.2. Concrete Structures and Components

As with the previous subsection, some of the same corrosive atmospheric environments could likely combine with one or more of the listed concrete degradation mechanisms (discussed in subsection 4.4.3.2) and lead to damage to the dry-storage system concrete or reinforced concrete foundation pad after only a few decades. This degradation is relatively easy to discover and correct.

The Safety Analysis Report produced by vendors for licensing each dry-storage system at a particular site presents the storage technical basis for the design license period. For storage out to 60 years, it is likely that the metal canister will not be breached, ensuring that the fuel and fission gases will remain contained.

Consequently, there is a strong technical basis and likelihood, that during a period of dry storage of 60 years duration, the fuel cladding and canister will not be breached, limiting degradation of the used fuel and maintaining containment of the used fuel and fission gases. Beyond 60 years, active corrosion mechanisms increase the likelihood that the canister may eventually be breached if inspection and maintenance programs are not put in place to identify and mitigate any problems found. This is expected to occur most rapidly at sites with a humid and warm sea climate, but it is uncertain when such a breach could occur and then how rapidly helium could be lost and be partially replaced with humid air.

6. Evaluation of Technical Bases for the Transportation of Used Fuel

The evaluation of the technical bases supporting safe transport of CSNF is carried out in a qualitative manner and using the requirements described in Section 3. In brief, Section 3 performance requirements specify that: (1) no release of used-fuel radioactive contents occurs to the environment, (2) dose rates on the external shipping cask surface are kept within prescribed levels, (3) an unsafe fuel configuration (accidental criticality) situation must be avoided, and (4) thermal performance must be maintained. Transport package manufacturers must demonstrate to the NRC that their package design meets the Section 4 requirements and test conditions as outlined in 10 CFR 71[31] and other regulatory requirements to be certified for use.

Section 4 of this report surveyed the existing state of knowledge regarding used fuel and a number of significant degradation processes that could impact used fuel stored for an extended period of time. A systematic and thorough understanding of the extended-storage behavior of all fuel (burnup) and cladding types is a prerequisite to analyzing the performance of fuel during handling and transport. This information is critical to obtaining NRC certification for transport, determining how ISG-11 on cladding considerations will be applied, and for guiding cask designers and owners in any additional design features that may be required to obtain package design approvals over the long term.

Shipping regulations are largely focused on the integrity of the transportation casks, which contain the used fuel, and maintaining subcriticality of the fuel. An important goal is to assure that in the event of transportation accidents the cask does not fail (radioactive material is not released). Besides the need to maintain containment and subcritical geometry of the fuel, applicants for transport certification must demonstrate that adequate surface temperature and shielding levels are met during transportation. Under accident conditions, some criteria may be less restrictive, since damaged fuel as a result of the accident could be expected, but analysis is still needed to comply with the regulations. For details as to specific requirements refer to Section 3.

A variety of truck and rail transportation package designs have been developed to meet the current federal requirements for packages. Most designs are a variation on a hollow tube with an impact limiter (absorb accidental impact forces) built into a sealed end and one or two closure lids designed at the other end. The tube design typically consists of layers of steel for strength; lead, depleted uranium, or concrete for gamma radiation shielding; and water, borated polymers, or concrete for neutron shielding. Nuclear utilities expect that used fuel currently stored in multi-purpose containers will not have to be repackaged and further that transport cask is sufficiently robust to protect against postulated terrorist assaults and accident scenarios.

A good summary of the transportation investigations that have been carried out in the United States and abroad, through 2005, is described in National Research Council report, *Going the Distance? The Safe Transport of Spent Nuclear Fuel and High-Level Radioactive Waste in the United States*.[187] In the report's analysis of the transportation safety risk, a principal finding was that the "committee could identify no fundamental technical barriers to the safe transport of used nuclear fuel and high-level radioactive waste in the United States. Transport by highway (for tens of tons) and rail (for hundreds to thousands of tons of used fuel) is, from a technical viewpoint, a low-radiological risk activity with manageable safety, health, and environmental consequences when conducted in strict adherence to existing regulations."[188] However, one caveat is that the study did not consider an extended storage period over which degradation mechanisms may significantly affect material properties and characteristics.

The report highlights a number of important points.[187] The hypothetical accident condition of a 9-meter free-drop test onto an essentially unyielding surface is a more severe mechanical loading test than any of the full-scale crash tests that have been carried out to date. This drop test imposes mechanical forces that are severe enough to bound any credible real-life accident conditions. Accident reconstruction and modeling studies described in the report suggest that there may be a very small number of credible accident conditions involving very long duration, fully engulfing fires that are potentially capable of releasing radioactive material from the package. The probability and consequences of such a release are not fully understood at present and research is needed into the effects of long-duration engulfing fires.

The report further recommends that limited, full-scale testing be conducted to validate modeling work as to how transport packages will perform under both regulatory and credible extraregulatory conditions. In addition, the report endorses the approach that integrates full-scale testing, scale-model testing, and finite element and associated structural analysis methods to demonstrate compliance with regulatory standards, noting that scale-model testing has some important limitations. It notes that a small number of full-scale package tests were carried out under various accident conditions in both the United States and the United Kingdom during the 1970s and 1980s.[189] Some of these tests resulted in small leaks being detected. A proposed new NRC test has not been conducted yet that would consist of a full-scale rail package (containing surrogate fuel elements) mounted on a rail carrier car placed at 90 degrees to a simulated rail crossing. The package would be subjected to a collision with a locomotive and several freight cars traveling at 60 miles per hour, followed by a fully engulfing, optically dense, fire for a duration of 30 minutes.[189] They recommend testing one full-size transport cask representative of the new generation of larger size casks developed over the last decade.

The performance tests discussed above are carried out using newly constructed transportation packages and either containing no fuel rods or new surrogate fuel-rods. As a result, these tests may not reflect actual behavior of packages and the fuel rods inside, especially multi-purpose packages and fuel rods that have aged for decades. On the other hand, margins of safety employing conservative material parameters and "worst-case" assumptions regarding applied loads will mitigate in the other direction. In fact, the purpose of large safety margins is to cover the many uncertainties that are present in human assessment of such behavior and performance.

After carrying out a sequence of transport cask performance tests under the specified "hypothetical accident conditions, the licensee must assure that there is no significant cladding failure."[29] Searching the literature, a number of information gaps are apparent that are related to this topic. The available research and test results do not provide sufficient information to confidently correlate or predict the expected fuel-rod damage state and several key variables: the expected initial physical-state of the transported fuel-rods and assembly structure, effect of the fuel-assembly design and construction, effect of the transport-cask design, effect of transport-cask aging, impact speed, impact accelerations, and impact orientation. *It is recommended that full-scale* *experiments or equivalent testing and modeling be preformed to validate the predicted behavior of fuel rods under extreme accident conditions using actual used fuel-rods loaded with fuel pellets and fragments.* Post accident characterization of fuel pellets and cladding would reveal the degree of accident damage. If a gross breach of cladding occurs, the release of fission gases, condensable vapors, and size distribution of solid particles should be characterized. Appropriate uncertainty analyses and a probabilistic risk analysis of these results would be useful.

The release of significant quantities of radioactive gasses and particles into the environment from a CSNF transport package during a severe accident would occur only if the package and one or more of the fuel rods were breached. If no fuel rods were breached, but the transport package did breach, the only source of radioactive material release would be radioactive contamination (crud) that was formerly attached to the external surfaces of fuel assembly components. The level of radioactive contamination from this latter source would be small and probably insignificant.

To date, U.S. and foreign countries have transported nuclear fuel safely for several decades without a significant release of radioactive material. In the past, small releases and external surface contamination non-compliance incidents have occurred,[190] but current NRC and IAEA regulation prohibit procedures and features that led to these problems. However, the current shipping and handling experience of CSNF in the U.S. and overseas is limited with respect to: the volume, distances, and type of fuel shipped; the storage period of CSNF before transport (aging concerns); observational testing using actual CSNF and aged transport casks; and the range of possible handling and transport conditions that may occur. Furthermore, safety, in the National Research Council report refers to normal and off-normal accident conditions during normal transport operations; it excludes security risks from sabotage, attacks, and theft.[191] Transportation security is an important element of safety that requires further assessment, but is beyond the scope of this report.

In summary, current transportation package modeling and testing provides sufficient confidence that packages can survive most severe accident scenarios without an increase in external radiation to levels that would endanger emergency personnel or the public.[192] As-built transportation packages are extremely rugged. However, there are a few extreme, but credible, accident conditions involving locomotive impact, extreme fires, and multiple scenario accidents where sufficient uncertainty exists that a minor radioactive release cannot be ruled out. More important is the need for additional research to better characterize the performance of aged (over the periods of long-term and very long-term storage) and extended-burnup fuel performance under both normal and accident conditions of transport to build a technical basis for safe transport. Research and testing should be focused on validating numerical modeling and scale-model assumptions and approaches used to support meeting transportation performance requirements.

Because public confidence in transportation of CSNF is an important concern to communities where fuel will be transported, this may be an additional reason that full-scale testing of actual transport casks be considered and carried out under credible accident conditions, which is not currently required. Full-scale tests build more confidence in the public mind and are more transparent than numerical simulation research conducted by experts. Such tests would help put the public and politician minds to rest that the NRC licensed transport packages are sufficiently robust to ensure safe transport over a wide range of possible conditions. This systematic testing should be fully documented and made publicly available.

7. Research, Development, and Monitoring Needs

As discussed in Sections 5 and 6, there is insufficient technical information available to answer with confidence the question of how long CSNF can be stored safely in dry-storage systems and then safely handled and transported. Section 5 contrasted empirical and theoretical approaches of accomplishing engineering work. This section identifies a number of information gaps that can be addressed (to some degree) by research, monitoring, inspection, testing, and analysis. How to prioritize these needs will be determined by the balance of empirical and theoretical approaches relied upon in technical basis arguments. This study was conducted to take full account of the large uncertainties that play a part in technical analyses that attempt to predict complex behavior decades into the future.

The long record of engineering applications has demonstrated that scientific and engineering applications of theoretical modeling or empirical knowledge to predict future behavior often are not sufficient. It is important that conceptual knowledge and supporting assumptions be validated with actual observational knowledge, to the extent possible. Because of incomplete knowledge, assumptions made, limited data, simplified models of nature, and overconfidence, it is essential that all important assumptions and conceptual models underlying predictions be validated to justify confidence in the predictions.[193]

Based on this review, a number of research needs have been identified to solidify the technical basis for extended dry storage particularly for new high-burnup fuels where data is particularly lacking. A summary list is provided below in Table 9:

Table 9. Top Research Needs for Long-term Used-Fuel Storage

1. Understanding long-term changes in the mechanical cladding behavior and fuel cladding degradation mechanisms potentially active during extended dry storage, including those that will act on the materials introduced in the last few years for fabrication of high-burnup fuels
2. Understanding and modeling the time-dependent conditions that affect aging and degradation processes such as temperature profiles, in situ material stresses, quantity of residual water, and quantity of helium gas
3. Modeling of age-related degradation of metal canisters, casks, and internal componentss during extended dry storage
4. Inspection and monitoring of fuel and dry storage systems to verify the actual conditions and degradation behavior over time, including techniques for assuring the presence of helium cover gas
5. Verification of the predicted mechanical performance of fuel after extended dry storage during cask and container handling, normal transportation operations, fuel removal from casks and containers, off-normal occurrences and accident events
6. Design and demonstration of dry-transfer fuel systems for removal of fuel from casks and canisters following extended dry storage

Additionally, based on this review, the NRC regulations need to be clarified and reviewed relative to expectations for extended dry storage. Our review found the regulations to be confusing as to expectations, especially with respect to the need for fuel assembly retrievability. Because storage and transportation are linked, it would be helpful to review the regulations as an integrated set to ensure that they are based on a risk assessment for safety significance and consequence.

NRC is continually reviewing physical security and terrorist threats. These should also be reviewed on a risk-informed basis using potential consequence analysis instead of establishing requirements in isolation.

At this point, due to the lack of a clear nuclear waste management policy for the U.S., used fuel may be stored at reactor sites for a long time before being moved to another interim storage facility, disposal site or a used-fuel processing facility. It is important to implement a comprehensive research and development program to provide additional confidence that used fuel can be safely stored, transported and handled in the future.

Because of the variety of fuel and cladding types, dry-storage system designs, historical procedures, and plant site environments, a research program needs to be developed that envelopes the types of used fuel and storage systems proposed. It is unrealistic to test or inspect all fuel types and storage systems. What is needed is a program of inspection for the purpose of validating technical understanding of degradation mechanisms for low and high-burnup fuels over time to provide assurance that degradation mechanisms are within the range of that expected.

8. Conclusions

The technical information currently available, together with the experience gained to date in the dry storage of used fuel, demonstrates that used fuel can be safely stored in short term and then transported for additional storage, processing or repository disposal, at least for low burnup fuel. However, additional information is required in order to demonstrate, with similarly high confidence, that high burnup fuel can be safely transported and any type of used fuel can be stored in dry storage facilities for extended periods without the fuel degrading to the extent that it may not perform satisfactorily during continued storage and subsequent transportation.

However, the Board recommends a number of research and development programs be implemented to demonstrate that used fuel can be stored safely in dry storage facilities for extended periods. However, research alone will not be sufficient. Because the experience base for extended dry storage of used fuel is short and the credible degradation phenomena are several and not robustly predictable in a quantitative sense, an in-service inspection and aging maintenance program appears to be necessary to support extended dry storage of used nuclear fuel. The technical details of such an in-service inspection program will depend on the desired safety objectives of extended dry storage. Consequently, a practical engineering approach that is based on the observational method and periodic assessments will likely be required to provide an adequate safety basis in addition to what can be learned from targeted scientific investigations.

The regulations concerning dry storage of used fuel do not currently address storage for extended periods. There is also some inconsistency between the regulations that apply to dry storage and those that apply to transportation and it is unclear how to meet both sets of regulations. It would be helpful in managing extended dry storage of used fuel if the regulations were to be revised as an integrated set and based on a risk assessment for safety significance and consequence. In addition, the Board considers that the regulatory requirements related to physical security and terrorist threats should also be reviewed on a risk-informed basis using potential consequence analysis and integrated with the storage and transportation regulations.

At this point, the nuclear waste management policy of the United States is unclear, with the result that used fuel will be stored at reactor sites for longer than originally foreseen. It is thus essential that the appropriate research and development programs, and monitoring and inspection programs, are implemented as a matter of priority in order to demonstrate that used fuel can be safely stored for extended periods and then transported and handled as part of a future waste management program.

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Acronyms

ASTM	American Society for Testing and Materials
BWR	boiling water reactor
CFR	*Code of Federal Regulations*
CS	carbon steel
Cs	cesium
CRUD	Chalk River Unidentified Deposits
CSNF	commercial spent or used nuclear fuel
DHC	delayed hydride cracking
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
EPRI	Electric Power Research Institute
GWd/MTU	giga-Watt days per metric ton of uranium
IAEA	International Atomic Energy Agency
ISFSIs	Independent Spent Fuel Storage Installations
ISG	Interim Staff Guidance (NRC)
Kr	krypton
LWR	light-water reactor (light water is ordinary water, H_20)
MOX Fuel	mixed oxide (mixed oxides of uranium and plutonium)
MPC	multi-purpose canister
MTHM	metric tons of heavy metal, (mostly uranium and some plutonium)
MWTA	maximum wall thickness average
Ni	nickel
Np	neptunium
NUHOMS	Nutech Horizontal Modular Storage System
NUREG	NRC Regulatory Guide
NUREG/CR	NRC Regulatory Guide/Contractor Report
NWTRB	Nuclear Waste Technical Review Board
PCI	pellet-cladding interaction
PCT	peak cladding temperature
PRA	probabilistic risk assessment
PTFE	polytetrafluoroethylene
PWR	pressurized water reactor
SAR	safety analysis report (document type submitted to NRC)
SER	safety evaluation report (document submitted by NRC in response to a SAR)
SNF	spent or used nuclear fuel
Sr	strontium
SS	stainless steel
TAD	DOE Transport, Aging and Disposal canister system
UO_2	uranium dioxide
U.S.	The United States of America
NRC	U.S. Nuclear Regulatory Commission
Zn	zinc
Zr	zirconium

Glossary

Accident –An undesirable event; especially one that could potentially do damage or harm to a dry-storage system, a transport package or its contents.

Aging – The general process in which characteristics of a structure, system or component gradually change with time or use due to physical, chemical, and/or biological processes. A given aging process may reach a point of equilibrium where material degradation essentially stops. Aging that is associated with dry storage usually results in changes in physical and/or chemical material structure and degradation of the engineering values of the material's mechanical properties. The term aging may be qualified. For example, due to heat alone, thermal aging or aging caused by the action of an oxidizing agent is termed oxidative aging.

Boron – A chemical element that absorbs neutrons, thus controlling or stopping a nuclear chain reaction.

Burnup – The amount of energy released from a fuel assembly in a reactor in terms of the thermal power of the plant multiplied by the time of operation (e.g., Gigawatt-days), divided by the mass of initial fuel loading (metric ton of uranium), which results in an overall reduction of fuel assembly reactivity. Low-burnup fuel is defined as a fuel pellet with an average cross-section burnup or assemblies with average assembly burnup of less than or equal to 45 GWd/MTU. High-burnup fuel consists of a fuel pellet with an average cross-section burnup or assemblies with average assembly burnup that exceeds 45 GWd/MTU.

Canister – A thin metal cylinder (typically an alloy steel about 0.5 in. thick) that is used to perform the functions of confinement.

Confinement – Those systems of a dry cask storage system that act as barriers to prevent the release of radioactive substances into the environment.

Containment – The ability of a component within the dry cask storage system to contain the CSNF in an acceptable geometry.

Degradation – Chemical and physical changes in a material that result in undesirable changes in the values of in-use properties of the material.

Normal – Expected radioactive wastes, operations and/or processes.

Off-normal – Any combination of radioactive waste, operations or processes that are not expected during normal activities; usually associated with damaged or failed materials, equipment or processes.

Nuclear Fuel – Fissionable material that was originally enriched sufficiently to be able to support a self-sustaining fission chain reaction when used to fuel a nuclear reactor. Also nuclear fuel can refer to fuel rods that include both fissionable pellet-material and the metal cladding that encases a stack of fuel pellets.

Package – Refers to the cask and any other packaging together with its radioactive contents as presented for transport.

Retrievability – The ability to remove CSNF from storage without the release of radioactive materials to the environment or radiation exposures in excess of the specified limits.

Spent Nuclear Fuel – Nuclear fuel that has been irradiated or "burned" in the core of a nuclear reactor or recycled one or more times, and implies the fuel cannot be recycled further. Note that this relatively new definition (in conjunction with the definition of used nuclear fuel) is different from the traditional NRC definition of spent nuclear fuel: "means fuel that has been withdrawn from a nuclear reactor following irradiation, has undergone at least one year's decay since being used as a source of energy in a power reactor, and has not been chemically separated into its constituent elements by reprocessing."[6] (Note: The terms "used" and "spent" fuel are used interchangeably for the purposes of this report.)

Used Nuclear Fuel – Nuclear fuel that has been irradiated or "burned" in the core of a nuclear reactor such that it can no longer effectively sustain a chain reaction or is no longer economically viable to "burn" in a light water reactor without treatment or reprocessing. (Note: The terms "used" and "spent" fuel are used interchangeably for the purposes of this report.)

Endnotes

[1] DOE FY 2010 Congressional Budget Request Volume 5, May 2009, DOE/CF-039, page 504.

[2] In the context of this report, *safety* refers to measures that protect used nuclear fuel storage facilities and transportation canisters against damage, failure, human error, or other accidents that could disperse radioactivity into the environment.

[3] In the context of this report, *security* refers to measures to protect used nuclear fuel storage facilities and transportation canisters against theft, sabotage, or terrorist attacks. Of course, whether a breach of a dry cask occurs due to a security event, accident, or natural event, all could lead to a release of radioactive materials – so in many respects a safety and security analysis overlap.

[4] The terms "used" and "spent" fuel are used interchangeably for the purposes of this report. The more recent definition suggests that "used" fuel is that fuel that comes out of light water reactors, which may be worthwhile to reprocess in the future. "Spent" fuel suggests that it will no longer be reprocessed to extract useable fuel in the form of uranium and plutonium for further recycling, but could be reprocessed as a waste management strategy or directly disposed.

[5] J. Bruno and R.C. Ewing (2006). "Spent Nuclear Fuel," *Elements*, Vol. 2, pp. 343-349.

[6] NRC (U.S. Nuclear Regulatory Commission, 2008). Title 10, *Code of Federal Regulations*, Part 72, "Energy: Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," Washington, DC: Office of Federal Register National Archives and Records Administration, U.S. Government Printing Office, as amended June 9, 2008 (10 CFR 72).

[7] IAEA-TECDOC-1293 (2002), "Long term storage of spent nuclear fuel – survey and recommendations," p. 41. The following eight issues are addressed during the current renewal process: (1) The intended functions of the pertinent systems, structures, and components (SSCs) of the ISFSI; (2) Degradation mechanisms and effects (aging) for specific SSCs; (3) Existing maintenance and monitoring programs and proposed modifications thereto; (4) Changes in environmental factors and conditions for the ISFSI; (5) Realised and predicted changes in site-specific characteristics; (6) Operational experience at this ISFSI and at others with similar or identical cask designs; (7) Applicable reported or non-reported ISFSI or cask events; and (8) Realised and projected decreases in thermal loading and source term for loaded casks.

[8] Storage canisters typically hold 32 assemblies of spent fuel rods from pressurized water reactors and 68 assemblies from boiling water reactors. These metal canisters are the key component around which dry storage systems are designed. The term heavy metal refers to actinides (the 14 chemical elements with atomic numbers from 90 to 103) including uranium, plutonium, and thorium.

[9] This figure is based on Figure 44 from NRC's Webpage, "Locations of Independent Spent Fuel Storage Installations" available at: http://www.nrc.gov/waste/spent-fuel-storage/locations.html

[10] Provided a new surveillance and maintenance program was initiated ("periodic inspections of areas not visible or readily accessible, verification of adequate aging management, and radiation monitoring activities to verify the effectiveness of the polymer neutron shield materials") and initiate a review of the technical basis and implications for future rulemaking regarding extended license and license-renewal periods. (NRC Staff, Surry License renewal, SECY-04-0175)

[11] 74FR47126-, "10 CFR Part 72 License and Certificate of Compliance Terms", September 15, 2009.

[12] IAEA-TECDOC-1293 (2002), p. 9.

[13] National Research Council of the National Academies (2006), *Safety and Security of Commercial Spent Nuclear Fuel Storage*, National Academies Press, Washington D.C., pp. 62-63. Most of the content of this paragraph and the following paragraph come from the information provided in this report.

[14] Listed at NRC's website, "Dry Spent Fuel Storage Designs: NRC Approved for General Use" available at: http://www.nrc.gov/waste/spent-fuel-storage/designs.html (as of August 2009)

[15] Rod McCullum, Nuclear Energy Institute, "Integrated Used Fuel Management Industry Perspectives," Presentation at NWTRB meeting, June 11, 2009, www.nwtrb.gov.

[16] NWTRB (2010), Dry Storage Tables_012309.xls and "Commercial Spent Nuclear Fuel Extended Dry Storage: Capacity Requirements Without A Repository Or Reprocessing." Projections made with NWTRB NUWASTE tool. This assumes a storage canister capacity of 32 Boiling Water Reactor assemblies and 69 Pressurized Water Reactor assemblies.

[17] NRC (2008). ADAMS ML# TBD

[18] The operation of placing damaged spent fuel in special containers is known as "canning." See Interim Staff Guidance - 1, Revision 2 Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function, Nuclear Regulatory Commission, Spent Fuel Project Office, May 11, 2007

[19] National Research Council of the National Academies (2006), p. 63 and Holtec International (2009), *Safety Analysis Report on the Hi-Storm FW System,* Marlton, NJ, Section 9.2.1 ("Overview of Loading Operations").

[20] Tom Brookmire (2009). "Data Needs for Long Term Storage of Used Fuel – Utility Perspective." Presentation for the Nuclear Waste Technical Review Board, June 11, 2009, Las Vegas, NV, Slide 3.

[21] The closure process involves welding a lid with thickness ranging from 4 to 10 inches (depending on design) to a canister shell that has a thickness of 0.5 to 1 inch. Although the weld dimensions are equal to or greater than the shell wall thickness, the weld does not penetrate the full lid thickness. Therefore, the canister lid to shell closure weld configuration is referred to as a partial-penetration and not a full-penetration weld.

[22] The diffusion coefficient of He through metal is reported to be far below 10^{-18} cm^2/s under dry storage temperature conditions (IAEA-TECDOC-1293 (2002), p. 61). Accordingly, over a period of 100 years, Helium diffusion through 1 mm of solid metal is practically impossible.

[23] NRC Interim Staff Guidance-11, Rev. 3 (2003). "Cladding Considerations for the Transportation and Storage of Spent Fuel." U.S. Nuclear Regulatory Commission, USA, Appendix A.

[24] NRC (2009). Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility Draft Report for Comment, NUREG-1536, Rev 1A, March 2009.

[25] NRC ISG-1 (2007). Interim Staff Guidance – 1, Rev. 2, "Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function," NRC Spent Fuel Project Office.

[26] NRC (1997). Standard Review Plan for Dry Cask Storage Systems: Final Report, NUREG-1536, January 1997. Also discussed in A. Machiels (2000), "Creep as the Limiting Mechanism for Spent Fuel Dry Storage," EPRI TR-1001207, December 2000, p. 1-1.

[27] NRC (1996). Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety, NUREG/CR-6407, INEL for NRC, February 1996.

[28] NRC ISG-2 (2007). Interim Staff Guidance – 2, Rev. 1, Draft "Fuel Retrievability," NRC Spent Fuel Project Office.

[29] NRC ISG-11 (2003). Interim Staff Guidance – 11, Rev. 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel," NRC Spent Fuel Project Office.

[30] The memorandum of understanding (MOU) between NRC and the Department of Transportation (DOT) was published in the Federal Register in July 1979 (FR Vol. 44, No. 128, 38690, 1979).

[31] NRC (2004). Title 10, *Code of Federal Regulations*, Part 71, "Packaging and Transportation of Radioactive Material," Washington, DC: Office of Federal Register National Archives and Records Administration, U.S. Government Printing Office, as amended Jan. 26, 2004, (10 CFR 71).

[32] Regulatory limits on radiation levels on the exterior surfaces of packages (10 CFR 71.47) restrict the age and burnup of spent fuel that can be carried in the package. The limits vary depending on whether normal or hypothetical accident conditions are assumed as explained in 10 CFR 71 sections 47 and 51. For normal transport conditions, the radiation level must not "exceed 2 mSv/h (200 mrem/h) at any point on the external surface of the package and the transport index does not exceed 10." The transport index is found by multiplying the maximum radiation level in mSv/h "at 1 m (3.3 ft) from the external surface of the package

by 100," which gives the maximum radiation level in mrem/h (10 CFR 71.4). If these radiation level limits are exceeded, the package may still be shipped by "exclusive use shipment only," provided the radiation level does not exceed 10 mSv/h (1 rem/h) on the external surface of the package, the shipment is made in a "closed transport vehicle" and is physically secured, and the radiation level anywhere on the vehicle does not exceed 2 mSv/h (200 mrem/h), nor 0.1 mSv/h (10 mrem/h) at any point 2 m (80 in) in a lateral direction from the vehicle, nor 0.02 mSv/h (2 mrem/h) in any normally occupied space. Under the hypothetical accident conditions, the radiation level must not exceed 10 mSv/h (1 rem/h) at any point 1 m (40 in) from the external surface of the package.

[33] For normal transport conditions, there shall be no loss or dispersal of radioactive contents "demonstrated to a sensitivity of 10^{-6} A_2 per hour, where A_2 is the maximum activity of the radioactive material calculated" according to 10 CFR Appendix A. Under the hypothetical accident conditions, there shall be no escape of krypton-85 "exceeding 10 A_2 in 1 week," and no escape of other radioactive contents "exceeding a total amount A_2 in 1 week."

[34] Under normal conditions of transport, contents must remain subcritical if water leaks in or out of the containment system assuming the most reactive credible configuration of the contents, water moderation to the most reactive credible extent, and full relection by water on all sides of the containment system. Under the hypothetical accident conditions, contents must remain subcritical per the same assumptions for normal transport plus considering the expected most credible configuration of contents consistent with the "damaged condition of the package and the chemical and physical form of the contents."

[35] For normal transport conditions, "no more than 5 percent reduction in the total effective volume of the packaging on which nuclear safety is assessed," "no more than 5 percent reduction in the effective spacing between the fissile contents and the outer surface of the packaging," and "no occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10 cm (4 in) cube." Under the hypothetical accident conditions, no specific packaging limits are given.

[36] Under normal transport conditions, in "still air at 38ºC (100ºF) and in the shade," no accessible surface of a transport package can exceed "50ºC (122ºF) in a nonexclusive use shipment, or 85ºC (185ºF) in an exclusive use shipment. An exclusive use shipment is made in a closed transport vehicle and is defined in footnote 32. The hypothetical accident conditions involve a high temperature thermal test as explained in 10 CFR 71.73.

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[47] CRWMS M&O (Civilian Radioactive Waste Management System Management and Operation Contractor, 2000). "Clad Degradation – Dry Unzipping," ANL-EBS-MD-000013 REV 00, Las Vegas, NV, ACC: MOL.20000503.0200, originator: Brady Hanson.

[48] ASTM C 1562 – 2003, "Standard Guide for Evaluation of Materials Used in Extended Service of Interim Spent Nuclear Fuel Dry Storage Systems," ASTM International, pp. 1-24.

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[168] Jun-ichi Tani, Masami Mayuzumi, Nobuyoshi Hara (2008). "Stress Corrosion Cracking of Stainless-steel Canister for Concrete Cask Storage of Spent Fuel." *J. of Nuclear Materials*, Vol. 379, 30 September 2008, pp. 42-47.

[169] NRC Information Notice 2009-26, *Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool*.

[170] W.L. Poe, Jr. (1998). *Final Long-Term Degradation of Concrete Facilities Presently Used for Storage of Spent Nuclear Fuel and High-Level Waste*, Revision 1, Tetra Tech NUS, Inc., Aiken, South Carolina, Section 2, "Degradation Mechanisms for Concrete and Reinforcing Steel."

[171] J. Connal and M. Berndt (2009). "Sustainable Bridges – 300 Year Design Life for Second Gateway Bridge," 7th Austroads Bridge Conference: Bridges Linking Communities, Auckland, New Zealand, 27 May 2009. Current standards for reinforced concrete structures subjected to severe environments (including bridges subjected to ocean water and climate) assume design life in the order of 100 years. Service life of 100 years has been used for major concrete bridge and tunnel projects in Europe since the early 1990s. These designs assume proper maintenance but no major repairs. AASHTO LRFD Bridge Design Specifications (2007) assumes a design life of 75 years. The Australian AS5100 Bridge Design standard adopts a design life of 100 years. The British BS5400-4 (1990) Code of Practice for Design of Concrete Bridges and Eurocode 2 (1992) adopts a design working life of 120 years. The noted reference above is based on a current design life of 300 years for a reinforced concrete bridge in a marine environment. The extended service life is obtained without great expense by using stainless steel reinforcing bars, thicker concrete cover to protect rebar, and improved performance quality concrete to minimize cracking. New types of concrete are now becoming available that can also extend service life such as bendable concrete (500 times more resistant to cracking) and ultra-high-density (UHD) concrete that greatly slows the rate of natural creep (increases durability).

[172] Poe (1998), Section 2.1.1 "Elevated Temperature," p. 2-2.

[173] Poe (1998), Section 2.1.2 "Freezing and Thawing," pp. 2-4 to 2-5.

[174] Poe (1998), Section S.3.1 "Surface Concrete Structures," p. S-1.

[175] Poe (1998), Section 2.1.3 "Leaching of Calcium Hydroxide," pp. 2-5 to 2-6.

[176] Poe (1998), Section 2.2.1 "Aggressive Chemical Attack," p. 2-6.

[177] L. Bertolini, B. Elsener, P. Pedeferri, and R.P. Polder (2004). *Corrosion of Steel in Concrete: Prevention, Diagnosis, Repair*, Wiley-VCH, 409 pp. Although the pH is a key factor influencing the chloride threshold for corrosion, several other factors can also be important such as the presence of microscopic air voids and cracks, temperature, the composition of the cement, and rebar surface roughness.

[178] Poe (1998), Section 2.2.2 "Corrosion of Reinforcing Steel," p. 2-7.

[179] Poe (1998), Section 2.2.3 "Reaction of Aggregates with Alkalines," p. 2-7.

[180] Poe (1998), Section 2.2.4 "Creep and Shrinkage," p. 2-8.

[181] Poe (1998), Section 2.2.6 "Irradiation," pp. 2-8 to 2-9.

[182] Electric Power Research Institute, "Feasibility of Direct Disposal of Dual-Purpose Canisters Options for Assuring Criticality Control," Report Number 101629, December 2008. Included in this report are calculations of the reactivity of two already-loaded dry storage systems located at the TVA Sequoyah reactor site. The calculations assume a fully flooded canister, no neutron absorbers except the neutron absorbers in the spent fuel itself. The calculations indicate that the spent fuel is (barely) subcritical.

[183] In rare cases, fuel is permanently removed from a reactor prematurely because the fuel has become damaged. Such fuel likely would have sufficient reactivity to sustain a nuclear reaction in a reactor environment. In addition, most of the fuel removed from a reactor at the end of the reactor's life would have sufficient reactivity to sustain a nuclear reaction in a reactor environment.

[184] Hugh A. Benton, "Approach to Repository Criticality Analysis," Presentation to the Nuclear Waste Technical Review Board, October 23, 1997, Las Vegas, NV, Slide 7.

[185] Thin films of liquid water, which are necessary for aqueous corrosion, begin to form when the relative humidity reaches ~70%. The exact figure depends on the material and its surface finish. This assumes no salt has been deposited on the canister surface. If salt has been deposited on the canister surface, the minimum relative humidity at which a water film begins to form is lower.

[186] J. Mitman (2002). "Dry Cask Storage Probabilistic Risk Assessment Scoping Study," EPRI Technical Report 1003011.

[187] National Research Council of the National Academies (2006), 339 pp.

[188] National Research Council of the National Academies (2006), pp. 7-8.

[189] National Research Council of the National Academies (2006), pp. 81-101.

[190] L.C. Cadwallader, S.J. Piet, S.O. Sheetz, D.H. McGuire, and W.B. Boore (2005). *Summary of Off-Normal Events in US Fuel Cycle Facilities for AFCI Applications*, Idaho National Laboratory for DOE, INL/EXT-04-02257, September 2005, pp. 60-63.

[191] National Research Council of the National Academies (2006), p. 29.

[192] National Research Council of the National Academies (2006), p. 174. "Estimated exposures in a maximally reasonably foreseeable accident would range from about 1100 person-rem for truck accidents to 9900 person-rem for train accidents. The maximally exposed individual is estimated to receive between 3 and 29 rem of radiation, which would be insufficient to cause acute radiation sickness or death. This exposure is estimated to produce between 0.5 and 5 latent cancer fatalities."

[193] Henry Petroski (1994). *Design Paradigms: Case Histories of Error and Judgment in Engineering*, Cambridge University Press, 221 pp. Provides many examples throughout history of how errors were introduced into the design process.