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Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases

September 1983

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Pacific Northwest Laboratory
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TECHNICAL BASIS FOR STORAGE OF ZIRCALOY-CLAD
SPENT FUEL IN INERT GASES

A. B. Johnson, Jr.
E. R. Gilbert

September 1983

Prepared for
the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830

Pacific Northwest Laboratory
Richland, Washington 99352

GLOSSARY

AECL - Atomic Energy of Canada Ltd.
ALARA - as low as reasonably achievable
ANL - Argonne National Laboratory, Argonne, Illinois
AR - at reactor (spent fuel storage)
atm - atmosphere
BCL - Battelle Columbus Laboratories, Columbus, Ohio
BWR - boiling water reactor
°C - degrees Celsius
Ci - curie
cm - centimeter
DIORIT - Swiss test reactor
DOE - U.S. Department of Energy
DWK - Deutsche Gesellschaft fuer Wiederaufarbeitung von Kernbrennstoffen, FRG
E - elastic modulus
EMAD - engine maintenance assembly and disassembly hot cell facility, NTS
EOL - end of life
EPRI - Electric Power Research Institute, Palo Alto, California
est - estimated
Fe - iron
FRG - Federal Republic of Germany (West Germany)
FTT - fuel temperature test (facility at EMAD)
g - gram
GCR - gas-cooled reactor
h - hour
H - hydrogen
He - helium
HEDL - Hanford Engineering Development Laboratory, operated by Westinghouse
Hanford Company
HTGR - high-temperature gas-cooled reactor (a type of GCR)
HWR - heavy-water reactor
I - iodine
IAEA - International Atomic Energy Agency, Vienna, Austria
ID - inside diameter
KKS - Kernkraftwerk Stade, FRG
Kr - krypton
KWO - Kernkraftwerk Obrigheim, FRG
KWU - Kraftwerk Union, Erlangen, FRG
KWW - Kernkraftwerk Wuergassen, FRG
LMFBR - liquid metal fast breeder reactor
LWR - light-water reactor
µm - micron
MeV - million electron volts
MPa - mega Pascal
MRS - monitored retrievable storage
MTU - metric tons uranium
MWd - megawatt day
Nb - niobium
NEA - Nuclear Energy Agency (part of OECD), Paris, France

Ni - nickel
 NRC - U.S. Nuclear Regulatory Commission
 NTS - Nevada Test Site, near Las Vegas, Nevada
 NWPA - Nuclear Waste Policy Act (of 1982)
 OD - outside diameter
 OECD - Organization for Economic Cooperation and Development, Paris, France
 OH - Ontario Hydro
 p - pico
 PCI - pellet-cladding interaction
 PHWR - pressurized heavy-water reactor
 PNL - Pacific Northwest Laboratory, Richland, Washington
 ppm - parts per million
 psi - pounds per square inch
 PWR - pressurized water reactor
 R - rad
 R&D - research and development
 RT - room temperature (~25°C)
 s - second
 SCAP - Solicitation for Cooperative Agreement Proposal
 SEAF - strain energy absorption to failure
 σ - stress
 Sn - tin
 SS - stainless steel
 SCC - stress corrosion cracking
 T - temperature
 tU - tonnes uranium
 TVA - Tennessee Valley Authority
 U.S. - United States.(also USA)
 W - watt
 WAK - Wiederaufarbeitungsanlage Karlsruhe, FRG
 WHC - Westinghouse Hanford Company, Richland, Washington
 WNRE - Whiteshell Nuclear Research Establishment, Pinewa, Manitoba, Canada
 wppm - parts per million by weight
 WR-1 - Whiteshell Reactor 1; heterogeneous, uranium, and organic-cooled
 WTSD - Waste Technology Services Division, Westinghouse Electric Company
 yr - year
 Zr - zirconium
 Zr-2.5 Nb - a zirconium-based alloy containing 2.5 wt% niobium
 Zry - Zircaloy

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ABSTRACT

This report summarizes the technical bases to establish safe conditions for dry storage of Zircaloy-clad fuel. Dry storage of fuel with zirconium alloy cladding has been licensed in Canada, the Federal Republic of Germany, and Switzerland. In addition, dry storage demonstrations, hot cell tests, and modeling have been conducted using Zircaloy-clad fuel. The demonstrations have included irradiated boiling water reactor, pressurized heavy-water reactor, and pressurized water reactor (PWR) fuel assemblies. Irradiated fuel has been emplaced in and retrieved from metal casks, dry wells, silos, and a vault. Dry storage tests and demonstrations have involved ~15,000 fuel rods, and ~5600 rods have been monitored during dry storage in inert gases with maximum cladding temperatures ranging from 50 to 570°C. Although some tests and demonstrations are still in progress, there is currently no evidence that any rods exposed to inert gases have failed (one PWR rod exposed to an air cover gas failed at ~270°C).

Based on this favorable experience, it is concluded that there is sufficient information on fuel rod behavior, storage conditions, and potential cladding failure mechanisms to support licensing of dry storage in the United States. This licensing position includes a requirement for inert cover gases and a maximum cladding temperature guideline of 380°C for Zircaloy-clad fuel. Using an inert cover gas assures that even if fuel with cladding defects were placed in dry storage, or if defects develop during storage, the defects would not propagate. Tests and demonstrations involving Zircaloy-clad rods and assemblies with maximum cladding temperatures above 400°C are in progress. When the results from these tests have been evaluated, the viability of higher temperature limits should be examined. Acceptable conditions for storage in air and dry storage of consolidated fuel are issues yet to be resolved.

EXECUTIVE SUMMARY

Wet storage is currently the only licensed interim storage option for spent light-water reactor (LWR) fuel in the United States. Dry storage offers an alternative that appears to be cost effective and technically sound, and it could be implemented on a modular basis to meet near-term storage shortfalls. This report examines the bases to select safe dry storage conditions and emphasizes the evidence for satisfactory cladding integrity.

The world-wide dry storage technology for Zircaloy-clad fuel was assessed, revealing several aspects that are relevant to licensing in the United States:

- Dry storage of Zircaloy-clad fuel in nonoxidizing atmospheres has been demonstrated and licensed in Canada, the Federal Republic of Germany (FRG), and Switzerland. Canadian and Swiss cladding temperatures are relatively low, but FRG dry storage demonstrations have been conducted at maximum temperatures near 400°C. Maximum storage periods range from 1.5 years for FRG demonstrations to 8 years for Canadian fuel. Dry storage tests have been conducted in inert gases up to 570°C without causing rod failures.
- Zircaloy-clad fuel has been emplaced in inert atmospheres and monitored in dry wells, metal casks, concrete silos, and vaults; casks have been emplaced under both wet and dry conditions; fuel retrieval from all of the concepts has also been demonstrated; and both canistered and uncanistered fuel has been emplaced and retrieved. The operations have been conducted with no apparent damage to the fuel.
- A relatively small fraction of U.S. spent fuel assemblies (estimated to be less than 20%) is expected to enter interim dry storage before the federal government begins to accept fuel from utilities (1998). However, that fraction could involve over 20,000 assemblies.
- A large fraction of the U.S. spent fuel inventory has low decay heats, suggesting that fuel can be selected to achieve relatively low storage temperatures, at least for unconsolidated fuel.

- The development of small cladding defects under dry storage conditions cannot be fully ruled out, but dry storage regimes have been identified where cladding defects are not expected to substantially impact fuel storage and retrieval operations.
- Due to thermal gradients in the storage container, only a small fraction of the rods will be at the hottest temperature. In fact, a significant length of each rod will be at temperatures below the maximum temperature. Thus, rod internal pressures that are calculated based on maximum rod temperatures will be conservative.
- Internal pressures will decrease as temperatures decline throughout the storage period, resulting in even more conservative fuel cladding conditions.
- Tests of irradiated fuel with cladding defects in argon at 230°C and 325°C confirmed that storage in inert gas will preclude secondary mechanisms that could propagate cladding defects.

Dry storage demonstrations, hot cell and laboratory tests, Zircaloy materials characterization, and fuel rod modeling have been conducted to assess cladding integrity in dry storage. The following aspects summarize the work that has been completed:

- dry storage demonstrations - Thirty-nine irradiated LWR Zircaloy-clad assemblies^(a) containing ~5600 rods have been monitored for up to 5 years. Initial maximum cladding temperatures for these 39 assemblies ranged from 120 to 430°C. Lower temperature demonstrations (120°C) have operated for up to 8 years; demonstrations at higher maximum cladding temperatures (380°C), for up to 1.5 years. No cladding failures have developed in any rods stored in inert cover gases. To date, only one rod has shown evidence of failure; the rod was stored in an air cover gas at ~270°C. Zircaloy-clad pressurized heavy-water reactor fuel rods exposed at up to ~50°C showed no evidence of cladding degradation after 4 years in helium.

(a) 16 boiling water reactor assemblies (BWR) and 23 pressurized water reactor (PWR) assemblies.

- hot cell and laboratory tests - Investigations have been conducted on monitored irradiated Zircaloy-clad rods over a range of dry storage conditions for up to 1 year. Approximately 75 irradiated rods or rod segments have been exposed in inert cover gases from 100 to 570°C; no cladding defects have been detected. In other tests, rods were driven to failure; temperatures of 765 to 800°C were required before the rods ruptured. At slow ramp rates, the cladding failures were pinholes that were not large enough for escape of significant fuel particles. These forced ruptures occurred at temperatures that exceed foreseeable dry storage cladding temperatures, even in abnormal storage events.
- Zircaloy materials behavior technology - Much information has been published regarding corrosion, hydriding, radiation damage, creep, and mechanical properties. Assessments of published data provide the basis to model and project cladding behavior in dry storage.
- theoretical treatments and fuel behavior modeling - Peak cladding temperatures up to 430°C (depending on the character of the fuel) are predicted to be acceptable for inert gas storage conditions. Frequently, the temperatures derived from modeling efforts assume isothermal conditions over extended periods (40 to 100 years). Limits derived under this assumption are highly conservative with respect to actual dry storage temperatures, which decrease significantly in a few years.

Fuel integrity results obtained in a given dry storage facility are equally valid for other facilities and concepts operated under similar conditions. Therefore, results reported here are applicable to storage of Zircaloy-clad fuel in metal casks, dry wells, silos, and vaults under equivalent conditions.

Cladding integrity considerations require two principal specifications for dry storage: cover gas selection and maximum cladding temperature.

Cover Gas Selection. Cover gas options include: inert gas (helium or argon); low reactivity gas (nitrogen); or oxidizing gas (air or CO₂). Dry storage in oxidizing cover gases has not been fully defined and requires further investigation. In Germany, two PWR assemblies were stored in moist nitrogen;^(a) the assemblies did not show evidence of degradation. The large majority of dry storage experience with Zircaloy-clad fuel involves inert cover gases. Storage in inert gases is the most conservative approach because these gases do not react with the cladding or with exposed fuel if cladding defects are present. Storage conditions in nitrogen appear to deviate only slightly from storage conditions in inert gases.

Maximum Cladding Temperature. Because no cladding failures have occurred in Zircaloy rods stored in inert cover gases, there is no success/failure boundary to serve as a clear selection criterion. Selection of a conservative cladding temperature guideline is therefore based on the following considerations:

- A dry storage demonstration with irradiated fuel assemblies at a maximum cladding temperature of 430°C has operated since February 1983; another demonstration at 385°C has operated since February 1982, and the temperatures have now declined below 250°C.
- Initial modeling assessments have recommended a maximum temperature of 380°C for Zircaloy-clad fuel for storage times up to 100 years.
- The experience base suggests that dry storage at 380°C has ample safety margins. No rods have failed in inert gas exposures up to 570°C, and rods forced to failure required temperatures from 765 to 800°C to produce ruptures.
- Tests and demonstrations are also being conducted at maximum cladding temperatures of 400 to 450°C. After these evaluations are completed, the 380°C guideline should be re-evaluated to determine if it is possible to justify a higher guideline temperature.

(a) One assembly has been stored for ~1 year.

- Storing fuel with cladding defects in inert cover gases is still considered conservative because the defects cannot propagate. However, reasonable fuel screening techniques should be used to minimize the size and number of cladding defects, particularly larger defects where loss of fuel pellets or particles during fuel handling is a possibility.

Due to low heat ratings, older, unconsolidated U.S. fuel is expected to have cladding temperatures considerably below 380°C in dry storage. However, relatively short temperature excursions during cask drying operations suggest a need for temperature margins. Also, allowable temperatures during drying operations may deserve special treatment because the above-normal storage temperature durations are relatively short.

Evidence presented in this report provides a basis to license dry storage of Zircaloy-clad fuel in inert gas at 380°C. Further investigations may be required in the following areas, depending on utility and federal dry storage needs and directions:

- Satisfactory regimes for LWR fuel storage in air could provide a storage option with minimal monitoring requirements.
- Dry storage of consolidated LWR fuel may be demonstrated and evaluated; temperature limits for older consolidated fuel must be considered in the context that temperatures decline less rapidly than for more recently discharged fuel.
- The types of mobile radioactive species and inventories and their impacts during foreseeable fuel recovery and handling operations should be further defined.
- Storage system integrity is a significant consideration but has site-specific and system-specific aspects that are not considered in detail in this report.

- Tests and demonstrations are planned and under way that will augment the statistical data base for 1) longer storage times and 2) temperatures above 400°C over relatively short time spans. Operation at temperatures above 400°C for short periods may be relevant to some cask drying operations or for consolidated or recently discharged fuel.

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1.0 INTRODUCTION

Many U.S. utilities have shown interest in developing dry storage of spent fuel to a licensed status, offering an alternative to wet storage for interim spent fuel management. Dry storage of Zircaloy-clad fuel is licensed in several countries, and there is evidence to support licensing in the United States. However, near-term U.S. licensing will probably take a conservative position while outstanding issues such as storage in air are being resolved.

This study was conducted by Pacific Northwest Laboratory (PNL)^(a) to update a prior comprehensive assessment of Zircaloy-clad fuel behavior in dry storage.⁽¹⁾ In this report, the status of dry storage technology is summarized. The methods and scope of surveillance to define the integrity of fuel cladding in dry storage tests and demonstrations are discussed. The status of efforts to model Zircaloy-clad fuel behavior in dry storage is summarized. The types of reactor-induced cladding defects that may enter dry storage are discussed and illustrated. The types of defects that could develop during dry storage are considered, and the impacts of cladding defects during interim storage and fuel retrieval operations are discussed.

Light-water reactors (LWRs) have either Zircaloy or stainless steel (SS) fuel cladding. Only ~5% of the U.S. LWR fuel inventory in wet storage has SS cladding, and there is no indication that SS-clad fuel will be placed in dry storage in the near term. Therefore, only Zircaloy-clad fuel behavior is addressed in this report.

(a) Operated for the U.S. Department of Energy (DOE) by Battelle Memorial Institute.

2.0 STATUS OF DRY STORAGE TECHNOLOGY

Dry storage of Zircaloy-clad spent fuel has received operational licenses in Canada and Switzerland and demonstration licenses in the Federal Republic of Germany (FRG). Operational dry storage of LWR fuel was licensed at the FRG Gorleben site in September 1983, and storage in metal casks is expected to begin in 1984. Other types of spent fuel are also being stored under dry conditions in France, Japan, the FRG, the United Kingdom, and the United States. In the United States, there is a strong motivation to license dry storage of LWR fuel to augment the options available to electrical utilities to meet their interim fuel storage needs.

The main objective of this report is to assess the data base for Zircaloy-clad fuel integrity. That data base includes the following aspects:

- foreign and U.S. dry storage demonstrations involving irradiated Zircaloy-clad fuel assemblies⁽¹⁻⁶⁾
- foreign and U.S. dry storage research involving irradiated Zircaloy-clad fuel rods and rod segments⁽⁵⁻¹⁰⁾
- relevant technology from the published literature; for example, Zircaloy stress corrosion cracking (SCC) phenomena⁽¹¹⁻¹⁴⁾ and radiation damage effects⁽¹⁵⁾
- theoretical assessments and modeling of Zircaloy-clad spent fuel behavior in dry storage.^(10,14,16,17)

In addition to data that are directly relevant to dry storage of Zircaloy-clad fuel, there is other experience regarding dry storage and handling of irradiated fuel, including:

- dry shipments of all fuel types, including extended shipments of Zircaloy-clad fuel by sea from Japan to European reprocessing plants, for up to 3.7 months, at cladding temperatures up to 380°C;⁽¹⁾ FRG fuel has been shipped with cladding temperatures as high as 450°C

- 40 years of dry handling experience involving all fuel types in hot cells; Zircaloy-clad rods have been in dry storage for up to a decade at hot cells⁽¹⁾
- dry storage of gas-cooled reactor (GCR) and liquid metal fast breeder reactor (LMFBR) fuel in dry well and vault facilities that began in 1964.^(1,2)

The major dry storage facilities and the scope of their operations are summarized in Appendix A. Facilities to store GCR, LMFBR, boiling water reactor (BWR), pressurized water reactor (PWR), pressurized heavy-water reactor (PHWR), and organic-cooled reactor (WR-1) fuels are included. Facility types include metal casks, dry wells, concrete silos, and vaults. All facility types have been licensed.

2.1 EXPECTED SCOPE OF DRY STORAGE FOR U.S. LWR FUEL

Wet storage will almost certainly remain the principal U.S. interim LWR fuel storage technology for electrical utilities. All LWR fuel will continue to be discharged to water pools and will remain in wet storage until decay heats decrease to levels consistent with the maximum allowable cladding temperature in dry storage. Furthermore, the major fraction of the U.S. LWR inventory will probably remain in reactor pools for interim storage. Wet storage is a demonstrated, safe technology that has functioned effectively for 40 years. It is likely that utilities will fully utilize their investments in wet storage facilities and dry storage will be supplemental. Projections suggest that interim storage needs for U.S. utilities will exceed estimated wet storage capacities beginning in the period from 1984 to 1986 (Figure 1). The line referred to as "Additional Requirements" in Figure 1 relates to storage needs not met by currently projected wet storage capacities. Those storage needs could be met by options that include:

- increases in at-reactor (AR) capacity (probably by adding dry storage on a modular as-needed basis)
- federal interim storage - The Nuclear Waste Policy Act (NWPA) of 1982 provides for up to 1900 MTU of commercial spent fuel at federal

FIGURE 1. Distribution of Projected U.S. LWR Spent Fuel Inventory in Various Storage Modes. Based on data from References 18 and 19; composite of BWR and PWR assemblies.

sites; the Act does not specify whether the storage technology will be wet or dry. Utilities must meet stringent U.S. Nuclear Regulatory Commission (NRC) eligibility requirements to use federal interim storage.

- up to 300 MTU at federal facilities for dry storage research and development (R&D).

Reprocessing, while not precluded by current U.S. policy, has no prospects for implementation to relieve the storage situation over the next several years. The 1982 NWRPA specifies that the federal government will begin to accept fuel from utilities in 1998 for storage in monitored retrievable storage (MRS) or repository facilities, provided that utilities have not elected to reprocess. Until then, interim dry storage is expected to handle the storage needs not met by wet storage until MRS or repository options have been implemented.

2.2 STATUS OF U.S. LICENSING FOR DRY STORAGE OF LWR FUEL

Several U.S. utilities have indicated that in the period from 1984 to 1988 they will require additional storage capacities.⁽¹⁸⁾ Lead times for pool licensing and construction preclude meeting these storage needs in the required time frame with additional wet storage capacity, although wet storage options such as rod consolidation or trans-shipment may be licensed at some reactors. Favorable economic and technical considerations⁽²⁰⁾ have led to considerable utility interest in dry storage. DOE has taken steps to assist the utilities in implementing dry storage, including the following actions:

- sponsoring R&D programs to improve the technical basis to design and license facilities
- promoting international cooperation in dry storage to broaden the available data base
- participating with utilities in specific dry storage demonstrations; Solicitation for Cooperative Agreement Proposals (SCAPs) have developed responses from utilities; up to three demonstrations are allowed by the 1982 NWPA.

Prospects to license dry storage of Zircaloy-clad fuel in inert and air cover gases are briefly discussed below.

Inert Gas Storage. The first license applications for dry storage of Zircaloy-clad fuel in the U.S. will probably be based on relatively conservative conditions. Aspects of the conservative position will almost certainly involve selection of a nonoxidizing cover gas. The first fuel placed in dry storage will probably be unconsolidated; however, consolidated fuel may be placed in dry storage in the future.

The FRG licensing authority has issued demonstration and operational licenses for storage of Zircaloy-clad BWR and PWR fuel at temperatures in the vicinity of 400°C in helium. Confirmatory tests at 400 to 450°C are under way in the FRG.^(3,4,9,10) The U.S. position will factor in information from these FRG tests.

Licensed dry storage facilities in Canada, FRG, and Switzerland all utilize helium cover gas. Two FRG demonstrations with irradiated PWR assemblies were exposed to moist nitrogen. While nitrogen is not fully inert, it has low reactivity toward Zircaloy and toward fuel exposed at cladding defects.^(8,21)

Air Storage. Spent fuel stored in an air atmosphere would not require further monitoring to assure inertness. Oxidation of the Zircaloy cladding is not the central issue in air storage.⁽²¹⁾ Rather, the principal issue arises from the fact that exclusion of all reactor-induced cladding defects from dry storage cannot be assured. In addition, the prospects that cladding defects will develop during storage are not fully defined. The principal licensing consideration is to identify a temperature where substantial degradation of exposed UO_2 and surrounding cladding will not occur over the expected storage period. Degradation occurs when significant UO_2 oxidizes to U_3O_8 , resulting in expansion of the fuel. This expansion results in formation of fuel particulates, releases fission products, and can cause the cladding to crack.^(6,7,22,23,24) Studies are under way in several countries to define acceptable conditions to store spent oxide fuel in air.

2.3 FUEL INVENTORY CHARACTERISTICS AND CONSIDERATIONS

In the United States, no commercial fuel reprocessing has occurred since 1971; therefore, a substantial fraction of the stored fuel inventory consists of older fuel assemblies with low heat ratings, shown in Figure 2 as a function of reactor and cladding types. The distribution of PWR and BWR assemblies as a function of time in wet storage and cladding material is shown in Figure 3.

Correlations of cladding temperature and kilowatts per assembly are currently not highly precise: they depend on storage system design, and they are available for only a few systems. Maximum cladding temperatures measured in FRG CASTOR models 1B and 1C with 4 PWR and 16 BWR assemblies, respectively, are shown in Figure 4. Results from single-assembly demonstrations in the United States at the Nevada Test Site (NTS) in dry wells, silos, and vaults are also shown in Figure 4 for comparison. The correlations apply only to unconsolidated fuel. For cask storage, the data suggest that much of the U.S. LWR inventory shown in Figures 2 and 3 could be stored unconsolidated below 200°C.

FIGURE 2. Distribution of Decay Heats for U.S. LWR Spent Fuel Inventories Discharged Prior to January 1983 and Projected to 1986 (based on data from Reference 18)

FIGURE 3. Distribution of Wet Storage Times for U.S. LWR Spent Fuel Inventories Discharged Prior to January 1983 and Projected to 1986 (based on data from Reference 18)

FIGURE 4. Spent Fuel Peak Cladding Temperatures in Dry Storage Demonstrations as a Function of Decay Heat. CASTOR decay heats are based on ORIGEN calculations; NTS decay heats are based on calorimetry.

However, any cladding temperature estimates must be considered very preliminary and system specific. Large casks with increased fuel contents would be expected to have somewhat higher cladding temperature profiles than those shown in Figure 4. Utilities with near-term storage problems appear to have substantial inventories of older fuel. However, some utilities may request licensed dry storage of consolidated fuel, which would essentially double the kilowatts per storage position and result in higher cladding temperatures.

2.4 STORAGE SYSTEM CONFIGURATION

During reactor service and in wet storage and in most dry storage concepts, LWR fuel assemblies remain in a vertical position; however, horizontal storage modes are conceivable. Although fuel integrity is largely independent of storage mode (principally determined by storage temperature, rod internal pressure, and cladding conditions), it is justified to consider whether horizontal storage imposes significant mechanical conditions not encountered in vertical storage.

There are two operations where irradiated LWR fuel is placed in a horizontal position. During reactor discharge operations, PWR fuel moves from a vertical position to horizontal and back to vertical as it is transported from the reactor core to the spent fuel pool. No evidence has emerged suggesting that the fuel cladding is damaged significantly by these operations. The other, more significant operation is LWR fuel shipments in metal casks. Casks are loaded with one or more fuel assemblies underwater in the vertical mode; however, casks are shipped horizontally on trucks, rail cars, or ships. The fuel is subject to shocks due to handling and travel. However, the general perception is that damage to the fuel during shipping operations is hardly perceptible.⁽²⁵⁾

Approximately 1000 shipments (~2000 tU) of LWR fuel have been transported dry in Europe and from Japan to Europe; these shipments have included ~4550 BWR and PWR assemblies. Within the United States there have been ~1500 LWR fuel shipments (900 tU; ~4100 fuel assemblies). During sea shipments the fuel remains horizontal for up to 2 to 3 months, sometimes at cladding temperatures estimated at up to 385°C. Thus, the fuel resides horizontally for significant periods at conditions approaching those in dry storage without evidence of significant damage. If the fuel is not damaged during emplacement and retrieval, horizontal storage does not appear to offer a threat to cladding integrity.

On theoretical grounds there does not appear to be a basis to regard horizontal storage as a threat to the cladding. Cladding creep for Zircaloy-clad fuel at temperatures below 380°C is regarded as acceptable over time spans associated with temperature decay.⁽²⁶⁾ Fuel spacers and storage system walls restrain rods from substantial distortion. Temperature profiles differ some-

what for identical systems in horizontal and vertical attitudes, but a horizontal system that meets storage temperature guidelines is not expected to encounter cladding behavior that differs significantly from behavior in vertical modes.

2.5 CONTAINER DRYING OPERATIONS

When fuel is loaded into casks or canisters underwater, a drying operation is required to reduce cover gas moisture to acceptable levels. Vacuum drying appears to be a more effective method than purge drying. Evacuation of the container reduces heat transfer, resulting in elevated cladding temperatures. Based on experience, cask drying is expected to require several hours (2 to 5) to achieve a 1-mbar residual pressure.⁽²⁷⁾ FRG experience has shown that under relatively short drying cycles (2 to 5 h) temperature excursions above 380°C are unlikely. In the event that vacuum drying causes excursions above the cladding temperature guideline, two approaches are suggested:

- implement special procedures to keep temperatures below the guideline
- request a temperature variance for the relatively short span of the drying operation.

Procedures to keep cladding temperatures below guideline temperatures include partially resubmerging the cask in the loading pool after draining and closing the drains. Spraying water on the cask exterior would also minimize interior cask temperatures, and slow pump-down and periodic refills with helium would tend to minimize temperature excursions.

The basis to request a temperature variance for the short span of the drying operation lies in the fact that Zircaloy-clad fuel has been tested under dry conditions up to 570°C over time spans of many days with no cladding failures (see Tables 5 and 6 in Section 3.3). The absence of cladding failures suggests that the few hours required for drying will probably not compromise cladding integrity for fuel with cladding hoop stresses near or below those of the fuel types that have been tested. At temperatures near or above 400°C,

some annealing of radiation effects is likely to occur, even in a few hours. However, annealing results in higher cladding ductility and therefore tends to improve rather than degrade cladding integrity for subsequent storage.^(1,7,28)

Some cladding oxidation may occur due to reaction with moisture as drying proceeds. However, the moisture levels will rapidly dissipate during evacuation to levels where cladding oxidation will be insignificant. Temperatures above 600°C are required to cause substantial Zircaloy oxidation over a 1-day time span, even at high oxidant levels.^(1,21)

3.0 ZIRCALOY CLADDING INTEGRITY IN DRY STORAGE

In this section, relevant fuel characteristics are summarized, dry storage concepts and conditions are discussed, and the investigations of Zircaloy cladding integrity that have been conducted are described.

3.1 ZIRCALOY-CLAD FUEL CHARACTERISTICS

Zircaloy has survived for over two decades as the principal cladding for commercial water-cooled reactor fuel.^(a) A small fraction of fuel rods fails in the reactor, but zirconium alloy performance under reactor conditions (Table 1) has been sufficiently favorable to justify its almost exclusive use as cladding for BWR, PWR, PHWR, and selected test reactor^(b) fuels. The upper range of cladding temperatures during reactor service is similar to the proposed temperature limit for dry storage (380°C). Several computer codes have been developed to model the in-reactor behavior of Zircaloy-clad fuel; nonproprietary codes are outlined in Appendix B. Fuel rod internal pressures for a range of BWR and PWR conditions (based on GAPCON-2 calculations) are presented in Figures B.1 and B.2.

The characteristics of irradiated fuel placed in dry storage will be a combination of preirradiation factors, in-reactor effects, handling during discharge, shipping (if any), and in-pool handling and storage factors. The potential impacts of each phase of fuel history are summarized in Table 2; relevant experience is referenced. The principal conclusion from Table 2 is that preirradiation and in-reactor factors will probably have the major influences on LWR fuel assembly characteristics that are relevant for dry storage. Table 3 summarizes fuel characteristics that are most relevant to dry storage. A significant insight from Tables 2 and 3 and Appendix B is that irradiated LWR fuel has a wide range of characteristics, for example, end-of-life (EOL) pressures. Two factors contribute to relatively low EOL internal rod pressures for

(a) Only three operating U.S. LWRs and two operating foreign LWRs use SS cladding.

(b) For example, DIORIT and WR-1 test reactors.

TABLE 1. Comparison of Fuel Exposure Conditions (Reference 1)

Parameter	In-Reactor			Wet Storage	Dry Storage
	BWR	PWR	HWR		
Fuel centerline temperature, °C	1400 to 1800	1200 to 1700	1300 to 1700	100 ^(a)	To ~70°C above cladding temperature
Cladding ID temperature, °C	320 to 360	340 to 370	300 to 350	30 to 60 ^(a)	To ~10°C above storage temperature ^(b)
Pressure inside fuel rod, atm	5 to 50	40 to 150	25 to 100	(c)	1.7 x RT ^(d) pressure for 250°C 2.2 x RT pressure for 400°C
Pressure outside fuel rod, atm	70	140 to 160	70 to 110	up to 2	Subatmospheric
Fission gas released to fill gas, %	1 to 11	1 to 13	2 to 18	(e)	(e)
Typical assembly burnup, MWd/tU	27,000	35,000	7,000	Same as in-reactor	Same as in-reactor
OD heat flux, W/cm ²	70	100	100	0.03 ^(f)	0.03 ^(f)
Neutron flux (>1/MeV) n/cm ² -s	4 to 6 x 10 ¹³	5 to 6 x 10 ¹³	4 to 6 x 10 ¹³	~10 ⁴ to 10 ⁶	~10 ⁴ to 10 ⁶
Gamma field, R/h	~10 ⁹	~10 ⁹	~10 ⁹	~10 ^{5(f)}	~10 ^{5(f)}
Power changes	Substantial	Substantial	Substantial	Minimal	Negligible

(a) Nominal value shortly after discharge; depends on water temperature and decreases with the age of the spent fuel.

(b) Expected dry storage starting temperatures range from ~150 to 400°C, depending on concept and fuel type.

(c) Approximately one-half of the in-reactor pressure.

(d) RT - room temperature (25°C).

(e) Additional fission gas release during storage is negligible.

(f) Decreases with spent fuel age.

TABLE 2. Fuel Cladding Properties and Characteristics During Fuel Cycle

Phase	Cladding Properties or Characteristics	References
Preirradiation	Preirradiation cladding mechanical properties differ from vendor to vendor; however, these differences are largely pre-empted by radiation effects.	13
	Prepressurization of fuel rods with helium has occurred since the early 1970s for PWR fuel and since the late 1970s for BWR fuel. Gas pressures contribute to hoop stresses in dry storage.	1
In-reactor	Increased cladding strength due to irradiation.	13
	Decreased cladding ductility due to irradiation.	13
	Development of oxide and crud layers on rod surfaces.	1
	Ingress of hydrogen to bulk cladding.	1
	Increased rod internal pressures due to fission gas release.	1, B.4, B.5, B.6(a)
	Development of fuel and cladding interactions due to a combination of fuel swelling and cladding creepdown.	1, B.1, B.9
	Development of cladding defects in a small fraction of rods.	1, 29
Postirradiation handling	A few assemblies have been dropped during discharge and fuel movements in pools, but impacts of handling operations on fuel assembly characteristics appear to be minimal.	30
Pool storage	Wet storage of Zircaloy-clad fuel for up to two decades has not caused detectable degradation, based on fuel examinations conducted in Canada, FRG, the United Kingdom, and the United States.	31
Fuel shipments	Systematic assessments of fuel shipping experience suggest that impacts on cladding characteristics are not major.	25, 29

(a) See Appendix B references, p. B.5.

TABLE 3. U.S. Zircaloy-Clad LWR Fuel Characteristics

Fuel Assembly Characteristics	Range of Values	Potential Significance to Dry Storage	References
Rod internal prepressurization	BWR fuel: 1 atm to mid-1970s; 4 to 6 atm since late 1970s PWR fuel: 1 atm to early 1970s; 20 to 35 atm since early 1970s	Contributes to end-of-life rod pressures (see next entry).	1
End-of-life rod internal pressures ^(a)	BWR fuel: 20 to 35 atm for early low-burnup fuel (to mid-1970s); 20 to 40 atm for fuel since late 1970s PWR fuel: 20 to 35 atm for early fuel (to early 1970s); 40 to 80 atm for fuel since late 1970s	Generate hoop stresses in cladding that are proportional to internal pressure and storage temperature; see Appendix B.	1
End-of-life cladding mechanical properties	YTS range: 500 to 800 MPa UTS range: 600 to 900 MPa Ductility range: 1 to 8%	Appear adequate for handling, shipping, and storage.	28
Zirconium oxide thicknesses	BWR fuel: 10 to 120 μm ; 30 μm nominal average PWR fuel: 3 to 60 μm ; 20 μm nominal average	Oxides are tenacious and not susceptible to spallation.	1
Fuel crud layers	BWR fuel: up to 130 μm PWR fuel: up to 70 μm	Crud layers are superficial to oxide layers and can develop the tendency to spall; impact in dry storage depends on fuel retrieval scenario.	32, 33
Hydrides in cladding and assembly components	BWR cladding: 20 to 150 ppm PWR cladding: 20 to 150 ppm	Hydride reorientation below the solubility temperature leads to reduced strain to failure.	1

(a) atm = MPa $\times$ 9.86. End-of-life pressures include initial fill gas and fission gas released from fuel pellets during reactor service.

older fuel: low fill gas pressures and lower burnups, resulting in minimal fission gas release. The range of EOL rod internal pressures for LWR fuel in dry storage demonstrations appears to be representative of the range expected for U.S. fuel assemblies that are candidates for dry storage in the near term.

3.2 DRY STORAGE CONCEPTS AND CONDITIONS INVOLVING ZIRCALOY-CLAD FUEL

Four dry storage concepts are under consideration.⁽¹⁾ The concepts and facilities where irradiated Zircaloy-clad fuel has been stored in each concept are summarized in Table 4. Zircaloy-clad fuel assemblies were sealed in metal canisters or baskets prior to storage in dry wells, silos, and vaults. Assemblies stored in metal casks were not placed in canisters. The decision to can the fuel should be judged on the merits of each case.

Fuel is stored in a cover gas at ~1-atm pressure inside a sealed cask (Figure 5) or canister. Decreasing decay heats result in decreasing temperature ramps. Substantial longitudinal and axial temperature gradients exist in multi-assembly casks (Figure 5). An example of a predicted diminishing temperature history is shown in Figure 6. The peak cladding temperature is predicted to fall from ~380°C to below 100°C in 10 years for PWR spent fuel stored in a metal cask. Radiation levels in dry storage are much lower than in the reactor (Table 1); however, the radiation levels may be high enough to cause radiolytic reaction where susceptible species exist. PWR assemblies have been exposed to moist nitrogen for up to ~1 year without evidence of an adverse effect on the assembly materials.⁽⁴⁾

Some seasonal temperature variations occur for dry storage modules exposed to climatic conditions, but thermal inertia suppresses nocturnal variations. Therefore, temperature fluctuations are relatively slow and moderate.

Dry storage tests should account for the actual storage environment when simulating specific dry storage conditions. It is important to recognize that cladding integrity information obtained in one storage concept is equally relevant to other concepts if the conditions are similar.

TABLE 4. Dry Storage Concepts Involving Zircaloy-Clad Fuel^(a)

Dry Storage Concept	Country	Site	Fuel/Cask Type	No. of Assemblies	Storage Period	Licensing Status
Metal cask	FRG	Wueriggassen	BWR/CASTOR 1C	16	1982 to 1984	Licensed demonstration ^(b)
	FRG	Karlsruhe	PWR/CASTOR 1B	4	1983 to 1985	Licensed demonstration ^(b)
	Switzerland	Wuerenlingen	Test reactor/ CASTOR 1C	(c)	1983 and continuing	Licensed operation
Concrete silo	Canada	Whiteshell	WR-1	138	1975 and continuing	Licensed operation
			PHWR	360	1976 and continuing	
	USA	EMAD/NTS	PWR	1	1978 to 1982	Unlicensed demonstration
Dry well:						
Surface	USA	EMAD/NTS	PWR	4 ^(d)	1979 and continuing	Unlicensed demonstration
Subsurface	USA	Climax/NTS	PWR	11	1980 to 1983	Unlicensed demonstration
Vault	USA	EMAD/NTS	PWR	Up to 13	1979 and continuing	Unlicensed demonstration

(a) See Glossary for definition of terms.

(b) A licensed 1500-MTU cask storage facility is scheduled to begin operation at Gorleben, FRG, in 1984.

(c) 350 four-rod clusters 2.13 m long, totaling 3 MTU.

(d) There are currently only three assemblies in the dry well.

FIGURE 5. Conditions in Metal Storage Casks

FIGURE 6. Calculated Peak Temperature History for PWR Spent Fuel in a CASTOR Metal Storage Cask (Reference 3)

3.3 INVESTIGATIONS OF ZIRCALOY CLADDING INTEGRITY IN DRY STORAGE

The basis to assess Zircaloy cladding behavior in dry storage comprises the following elements:

- dry storage tests and demonstrations with irradiated fuel where Zircaloy cladding integrity was determined by one or more surveillance methods
- Zircaloy material behavior tests
- theoretical modeling.

3.3.1 Dry Storage Tests and Demonstrations

Laboratory and Hot Cell Tests

Data from laboratory and hot cell tests on Zircaloy-clad fuel are summarized in Table 5. Time/temperature histories for whole-rod tests on irradiated Zircaloy-clad fuel in inert cover gases and air are summarized in Figure 7.

The tests spanned the following characteristics:

TABLE 5. Laboratory/Hot Cell Tests Involving Zircaloy-Clad Fuel Rods

Country	Organization	Cladding	Test Type	Temperature, °C	Cover Gas	Burnup Mwd/MTU	No. of Rods			Surveillance Methods and Results					References
							BWR	PWR	Test Time, days	Type ^(a)	Number ^(b)	Last Date	Results		
Canada	AECL/OH	Zircaloy-4	Rod capsules	220 to 250	Air	7,000 to 8,000	-	13 (PHWR)	30	3,4,6	-	In progress	Cladding defects elongated	22	
Canada	AECL/OH	Zircaloy-4	Whole rods	150	Moist air	5,000	-	8	1000	To be analyzed in 1984		-	In progress	5	
				150	Dry air	5,000	-	8	1300	To be analyzed in 1984			In progress		
FRG	KWU		Rod capsules	100 300	He + I ₂ He + I ₂	0 0	- -	- -	550	-	-	-	No failures	3,(d)	
FRG	KWU	Zircaloy-4	Rod capsules	350 to 450	Helium	To ~40,000	-	20	40 to 170	-	-	-	In progress	3,(d)	
FRG/Italy	Ispra/NUKEM	Zircaloy-2	Whole rods	400	Helium	PWR: 33,000	14	9	180 at 400°C	1-3	15	3/82	No cladding failures	9,(d)	
		Zircaloy-4		430	Helium	BWR: 24,000	14	9	150 at 430°C	1	15	2/83	No cladding failures		
				450	Helium		14	9	30 at 450°C	1	6	7/83	No cladding failures (in progress)		
FRG	KWU	Zircaloy-4	Whole rods	to 400	Helium		-	6	~180	1-3,7,8	-	In progress	-	(d)	
USA	BCL/HEDL	Zircaloy-4	Whole rods	480 ^(e)	Limited air, helium	28,000	- -	3 3	190 190	1-5	1	1979	No cladding failures	7	
				510	Limited air, helium	28,000	-	3 3	325 325	1-5	1		No cladding failures		
				570 ^(e)	Limited air, helium	28,000	- -	1 1	30, 73 30, 73	1-5	1		No cladding failures		
USA	BCL/HEDL	Zircaloy-4	Whole rods	325	Limited air, argon	28,000	- -	3 3	87 87	2-5 2-5	- -	1983 1983	No cladding failures No cladding failures		
USA	BCL/PNL	Zircaloy-2	Whole rods	325	Rplnd. air, argon	17,000 24,000	1 1	- -	87 87	1-5 1-5	~5 ~5	1983 1983	Defects propagated Defects did not propagate	8	
USA	PNL	Zircaloy-4	Whole rods ^(c)	Increasing ramp to rod rupture (765 to 800°C)	Limited air	0	-	3	1 to 2	2-4	-	1982	Pinhole defects at 765 to 800°C	34	
USA	EPRI/TVA/ HEDL/ANL	Zircaloy-4	Rod capsules	250 to 350	Air	28,000	-	48 specimens	In progress	-	-	-	In progress		
USA	NRC/HEDL/ EG&G	Zircaloy-2	Whole rods	230	Unlimited air, argon	12,000	4	-	248	1-3	-	8/83 ^(f)	1 cladding defect elongated	24	
		Zircaloy-4	Whole rods		Unlimited air, argon	To 31,000	-	4	248	1-3		8/83 ^(f)	No cladding failures	24	

- (a) 1 - fission gas analyses
2 - nondestructive rod tests
3 - visual inspection
(b) Number of cover gas samples analyzed.
(c) Unirradiated rods; however, under test conditions, radiation would have been fully annealed.
(d) Unpublished data supplied by J. Fleisch, DWK, Hannover, FRG.
(e) Isothermal for 190 days; then four rods were transferred to 570°C.
(f) Phase 2 is in progress.

FIGURE 7. Temperature History and Test Periods for Full-Length Irradiated Zircaloy-Clad Rods (2 PWR indicates that two PWR rods were involved in the test)

- cladding temperatures - 100 to 570°C
- fuel types - BWR, PWR, PHWR
- cladding types^(a) - Zircaloy-2 and Zircaloy-4
- cladding condition - intact and defective
- cover gases - air, argon, helium, and a mixture of helium and iodine
- test times - 30 to 1300^(b) days

(a) Zircaloy-2: Zr-1.5 Sn-0.15 Fe-0.1 Cr-0.05 Ni (nominal composition);
 Zircaloy-4: Zr-1.5 Sn-0.02 Fe-0.1 Cr-0.005 Ni (nominal composition).
 (b) To be evaluated in 1984.

- rod internal pressures - to 9.5 MPa (1360 psi) at room temperature;
to 16 MPa (2300 psi) at 325°C
- rod hoop stresses - to 140 MPa (20,000 psi) at 325°C.

The test results can be summarized as follows:

- No intact cladding failed during the tests, either in inert or limited air environments.
- There was evidence that radiation damage annealing occurred for rods tested at 480 to 570°C. Oxygen diffusion effects were also evident.⁽⁷⁾
- Ispra/NUKEM tests included early temperature plateaus at 100, 200, and 300°C, while approaching the 400°C test temperature; more recently, rod temperatures were increased to 430°C and 450°C.⁽⁴⁾
- Cladding defects did not propagate in an inert gas but did propagate in replenished air at 325°C.⁽⁸⁾ Cladding defect propagation has also been observed in air at lower temperatures.⁽²²⁻²⁴⁾
- Unirradiated Zircaloy cladding capsules did not fail when subjected to iodine vapor at 100°C and 300°C for test times to 550 days.⁽³⁾ Maximum iodine vapor concentrations (up to 1×10^{-5} g/cm²) exceeded the threshold concentrations for iodine-induced SCC by up to a factor of 10.⁽¹²⁾
- Unirradiated whole rods developed pinhole ruptures at 765 to 800°C in slow ramp tests. The fact that the rods were unirradiated appears to have minimal significance because the high test temperatures would cause radiation damage to anneal.⁽³⁴⁾ Rods were of standard PWR design, with ~400-psi fill gas pressures.

Dry Storage Demonstrations

Table 6 summarizes the results of dry storage demonstrations of irradiated fuel assemblies with zirconium alloy cladding. Time/temperature histories for dry storage demonstrations with irradiated Zircaloy-clad fuel are summarized in Figure 8. The temperatures declined on ramps determined by nuclear heat

TABLE 6. Dry Storage Demonstrations Involving Zircaloy-Clad Fuel

Country	Site	Fuel Type	Fuel Burnup, MWd/MTU	Cladding	Storage Concept	Time in Wet Storage	No. of Assemblies	Dry Storage Period	Cover Gas	Maximum Cladding Temperature, °C		No. of Rods	Integrity Verifications				References
										Initial	Current or Final		Type ^(a)	Number ^(b)	Date ^(c)	Results	
Canada	WMRE	WR-1	6,000	Zr-2.5 Nb	Silo	0.5 to 4 yr	+138	1975 +	Helium	120	-4	+2484	-	-	-	-	1,2,5
Canada	WMRE	PWR	6,000	Zircaloy	Silo	0.5 to 4 yr	360	1976 +	Helium	120	-4	6820	2-4	-	1982	No rod failures ^(d)	
FRG	KWO	PWR	33,000 to 43,700	Zircaloy-4	Test module	10 mo	1	60 days	Moist N ₂	300	250	180	1,5	3	1981	No rod failures	4,(e)
FRG	KWO	PWR	33,000	Zircaloy-4	Test module	4.5 mo	1	12 months	Moist N ₂	400	330	180	1-3,5	11	1982	No rod failures	4,(e)
FRG	KWW	BWR	28,000	Zircaloy-2	Metal cask	1 yr	16	2/82 to 2/84	Helium	385/180	250/120	896	1 ^(f)	9	5/83	No rod failures	27,35,(e)
FRG	KKS/ Karlsruhe	PWR	35,000	Zircaloy-4	Metal cask	10 mo	4	2/83 to 2/85	Helium	430/200	340/160	816	1,2,3	5	7/83	No rod failures	(e)
Switzerland	Wuerenlingen	Test reactor	17,000	Zircaloy-2	Metal cask	6 to 10 yr	3 MTU	May 1983 until final disposal	Helium	180 (est)	-	-	2	-	-	-	36,37
USA	EMAD/NTS ^(g)	PWR	28,000	Zircaloy-4	Surface dry well	2 to 3 yr	4	1/78 +	Helium	170	120	816	1-5 ^(h)	6	7/83	No rod failures	1,38
USA	EMAD/NTS ^(g)	PWR	28,000	Zircaloy-4	Silo	2 to 3 yr	1	11/78 to 8/82	Helium	150	100	204	1-5	1	6/82	No rod failures	1,38
USA	EMAD/NTS ^(g)	PWR	28,000	Zircaloy-4	Vault	2 to 3 yr	12	12/79 +	Helium	220	-	2448	1-5	5	7/83	No rod failures	1,38
USA	Climax/NTS	PWR	28,000	Zircaloy-4	Deep dry well	2 to 3 yr	11	To 4/83	Helium	230	~160	2244	1-5	11	7/83	No results to date	
USA	EMAD/NTS	PWR	28,000	Zircaloy-4	FTT	2 to 3 yr	1 ⁽ⁱ⁾	3/83	Air	275	260	202	1-5	7	9/83	One failed rod	(j)

(a) 1 - fission gas analyses
2 - nondestructive tests on rods
3 - visual inspection
4 - destructive analyses
5 - sipping.

(b) Number of cover gas samples analyzed.

(c) Date of last cover gas analysis.

(d) Rods in one basket were examined after ~4 years.

(e) Unpublished data supplied by J. Fleisch, DMK, Hannover, FRG.

(f) Precharacterization: sipping; profilometry; eddy-current; and oxide thickness measurements on 10 rods.

(g) See Appendix C for details of Turkey Point fuel assembly histories.

(h) One of 18 assemblies analyzed at hot cell; 5 assemblies at NTS have precharacterized rods; only interim visual examinations and fission gas analyses have been conducted since the NTS demonstrations began.

(i) Three PWR assemblies have been tested in the fuel temperature test (FTT) module, one at a time.

(j) Unpublished data supplied by A. R. Haki, WTSO, and DOE/NVO.

FIGURE 8. Temperature Ranges and Times for Storage Demonstrations Involving Zircaloy-Clad Fuel Assemblies. The hypothetical curves are based on 24 1-kW PWR assemblies in a metal cask.

decay.(a) In most cases, the demonstrations are continuing, and cladding integrity has been assessed by cover gas monitoring for fission gases (principally ^{85}Kr). Where cask fuel temperature gradients are reported (KWW and KKS), the upper and lower bounds are given in Table 6. The range of conditions in the demonstrations are summarized below:

(a) In the engine maintenance assembly and disassembly (EMAD) fuel temperature test (FTT), nuclear heat was augmented by electrical heat to reach the desired test temperature. The test temperature was programmed to decline on a prototypic ramp.

- temperatures - $-4^{(a)}$ to 430°C
- fuel types - BWR, PWR, PHWR, test reactor (WR-1 and DIORIT)
- cladding types - Zircaloy-2, Zircaloy-4, Zr-2.5 Nb
- cladding condition - no known cladding defects prior to dry storage^(b)
- cover gases - air (FTT), helium, moist nitrogen
- test times - up to 8 years (since 1975) for Canadian fuel at low temperatures (120°C) and continuing; since 1978-1979 for intermediate temperatures at EMAD (up to $\sim 230^{\circ}\text{C}$); up to 1.5 years in the FRG at higher starting temperatures (to 400°C)
- radiation levels - 10^4 to 10^6 R/h (estimated).

The results of the dry storage demonstrations on irradiated fuel assemblies can be summarized as follows:

- No rods have failed in inert or moist nitrogen cover gases; one rod failed at $\sim 270^{\circ}\text{C}$ in an air cover gas. No assessment of the failure has been made, but it is not likely that the air cover gas contributed to the failure because cladding oxidation rates were low in the test temperature range.⁽²¹⁾ Most cover gases involved in the demonstrations (Table 6) have been sampled several times to monitor for radioactivity release.^(c)
- Only $\sim 10\%$ of the KWW rods are estimated to be at the highest temperature (Table 6 shows the temperature range). Thus, within a given multi-assembly demonstration, the rods move through a relatively wide temperature range as the heat decays. Over the expected

(a) The decay heats of some Canadian fuel have decreased to levels where cladding temperatures are estimated to be below freezing in winter months.
 (b) EMAD PWR assembly B41 was suspected to have a defective rod; however, the assembly has resided only in a helium cover gas.
 (c) Cover gases frequently show levels of ^{85}Kr that are far less than the mobile inventory in a single rod. In the EMAD test, a single rod failure resulted in an increase in ^{85}Kr by 6 orders of magnitude.

2-year KWW test, rods will have resided at temperatures from $\sim 385^{\circ}\text{C}$ down to $\sim 100^{\circ}\text{C}$. Thus, questions regarding possible selective failure mechanisms in various thermal regimes appear to have been addressed. The hottest rods will take ~ 10 years to pass through the thermal regime that the cooler rods have already traversed.

- Radiolytic effects of moist nitrogen environments (for example, HNO_3 formation) did not appear to degrade the Zircaloy-clad fuel detectably in KWO tests.
- Collectively, the tests have involved almost 15,000 individual fuel rods over a wide range of thermal conditions. Approximately 9300 Canadian rods have not had cover gas monitoring; the remaining ~ 5600 rods have been monitored and only one rod showed evidence of failure, as indicated above.
- The hottest rods in the cask demonstrations ($>400^{\circ}\text{C}$) may have undergone substantial radiation damage annealing; annealing is expected to be minimal or insignificant on cooler rods. Thus, a range of cladding metallurgical conditions may be represented in hotter demonstrations.

Scope of Surveillance on Zircaloy-Clad Fuel in Dry Storage

Tables 5 and 6 indicate the scope of dry storage investigations on irradiated whole rods and irradiated assemblies. In most cases, rod integrity was monitored by sampling cover gases for radioactivity that would be released if one or more rods failed. Krypton-85 was the principal isotope monitored; in some cases, tritium was also monitored. The tables indicate that in most demonstrations gas samples were analyzed periodically. Use of internal proportional counting on krypton isotopes separated by gas chromatography (activated charcoal column) equipment permits monitoring radioactive ^{85}Kr concentrations as low as 10^{-2} pCi/cm³, which is ~ 7 orders of magnitude below the inventory corresponding to free ^{85}Kr released if one rod failed (based on measured free ^{85}Kr for Turkey Point rods). The free ^{85}Kr in each of five Turkey Point rods analyzed was 5×10^4 pCi/cm³ (see Rod G9 in Figure 9). The more sensitive of two measuring methods permitted detection of ^{85}Kr in cover gas samples from five Turkey Point assemblies that had been in a dry storage demonstration at

FIGURE 9. ⁸⁵Kr Increase in FTT Cover Gas for Assembly B02

EMAD. Low ⁸⁵Kr levels appeared in all five analyses but were 5 to 6 orders of magnitude below the ⁸⁵Kr inventory corresponding to one rod failure.

Tables 5 and 6 summarize dry storage tests and demonstrations on ~15,000 Zircaloy-clad rods. As indicated previously, the cover gases of ~9300 rods exposed in Canadian silos at temperatures below 120°C have not been monitored. Some 5600 rods have been monitored during dry storage; ~3500 of the rods were in demonstrations at NTS at temperatures below 230°C for test times up to ~5 years. Appendix C summarizes the histories of individual Turkey Point PWR assemblies at NTS. Four FRG tests and demonstrations (Ispra, KWO, KKS, KWW) have included ~1900 rods at peak cladding temperatures near 400°C. However, due to thermal gradients, only a fraction of the rods were at the maximum temperature. In cooling, the FRG rods have collectively traversed temperatures from 430°C down to 100°C. In the KWW test, 896 BWR rods with maximum starting

temperatures of 380°C have been in test since March 1982. Cover gas monitoring was last reported in June 1983, and the maximum cladding temperature was 250°C at that time.

One U.S. whole-rod test series was conducted at Battelle Columbus Laboratories (BCL) at 480 to 570°C on 14 Turkey Point rods (Figure 7). A more recent series was conducted at BCL at 325°C with five overpressurized Turkey Point rods. The Ispra whole-rod tests included both BWR and PWR rods. BWR rods with reactor-induced defects were tested at BCL at 325°C, with one rod in replenished air and one rod in argon.⁽⁸⁾

In the tests and demonstrations discussed above, rods monitored in inert cover gases have not shown any evidence of failure. Other surveillance techniques (sipping, nondestructive testing, etc.) have been applied on a selective basis; no cladding degradation in dry storage has been indicated. The FTT now under way at the EMAD facility with an air cover gas has shown evidence that one rod has failed. Turkey Point Assembly B02 resided in the EMAD silo test for ~4 years at <150°C in helium. The assembly was removed from the storage canister for visual inspection, crud sampling, and removal of two archive rods. The assembly was then placed in the FTT stand. The temperature was increased with electric heat to 275°C and was programmed on a decreasing ramp. A rod failure was indicated by a marked increase in ⁸⁵Kr (Figure 9) and helium (Figure 10) in the air cover gas between 40 and 70 days after the test began; the test temperature was ~270°C when failure occurred. The assembly has not yet been inspected, but it is not expected that the air cover gas influenced the failure, based on the low cladding oxidation rates in the test temperature range.^(1,21)

Time/Temperature Relationships

Figures 6 and 8 indicate a range of decreasing fuel temperature ramps, corresponding to decreases in fuel decay heats. For example, maximum cladding temperatures for BWR fuel (KWW) that has cooled for 1 year decreased by ~100°C in the first year of storage. Temperature decreases depend on fuel age; older fuel will decay more slowly than recently discharged fuel with the same characteristics. Hotter, unconsolidated fuel tends to remain at maximum temperatures for a relatively short time. For example, the KKS PWR fuel (Figure 8) started

FIGURE 10. Helium Increase in FTT Air Cover Gas for Assembly B02

at 430°C but remained above 400°C for only ~2 months and extrapolates to below 300°C in ~1 year. Thus, isothermal test times at 400°C and above have exceeded the times that fuel resides above 400°C on typical storage temperature ramps. The isothermal temperature test periods greatly exceeded the time required for cask drying operations, as indicated earlier. Dry storage modeling efforts are beginning to apply cumulative damage methods to assess the effects of decreasing temperature ramps on fuel cladding behavior.⁽¹⁶⁾ These methods, when fully implemented, will improve the traditional modeling approach, which restricts analysis to isothermal data only.

3.3.2 Zircaloy Materials Behavior

This section briefly summarizes the status of knowledge regarding Zircaloy properties and behavior relevant to dry storage regimes. Postulated cladding degradation and failure mechanisms for Zircaloy-clad fuel in dry storage, based on assessments by several investigators, are summarized in Table 7.(1,3,15-17,39-41)

TABLE 7. Summary of Postulated Cladding Failure Mechanisms in Dry Storage

Region in Figure 11	Mechanism	Relationship to Dry Storage	Known Consequences	Issues	References
A	Rapid stress rupture	Thermal excursions.	Major excursions generate minute defects under postulated dry storage temperature ramp.	What temperature excursions are credible?	17,26
		Mechanical overloads, for example, fuel handling events.	Has caused no significant impacts during fuel handling operations.	Are overload situations due to fuel handling in dry storage operations foreseeable?	17
B	Stress corrosion cracking	Cladding failures observed in reactor and test thermal regions similar to dry storage conditions.	Requires combination of aggressive fission products and high stresses (~ 180 MPa).	Are fission products aggressive after extended storage?	11-13,15,42-45
B	Strain rate embrittlement	Low ductility specimen failures observed in thermal regions similar to dry storage conditions.	No fission products were present.	Are required stresses and strain rates ever present in fuel under dry storage conditions?	40,41,46
C	Slow creep rupture	Decreasing dry storage temperatures suppress creep.	Observed only at atypical stresses in laboratory tests.	Will high-pressure, high-burnup fuel become a candidate for dry storage? Will localized stresses due to incipient defects lead to creep rupture?	15-17,40,41
			Cladding breaches are minute.		47
C	Cladding oxidation ^(a)	Insignificant in limited air or oxidizing atmospheres below 400°C.	Oxide films are coherent and protective.	None	1,21
C	Hydride reorientation	Dry storage temperatures eventually decrease below the hydrogen solubility limit.	Radially oriented hydrides reduce the strain for failure.	Do dry storage stresses and temperature conditions cause hydride reorientation?	1,48
		Hoop stresses may cause hydrides to become radially oriented.		If so, will cladding creep strains in excess of the failure levels be encountered during dry storage?	1,48
C	Fuel oxidation	Some interest for qualifying air storage.	UO ₂ oxidizes to low-density U ₃ O ₈ by oxygen ingress through cladding breaches and causes clad splitting; rates decrease with temperature.	Below what limiting temperature can monitoring of inert gas protection be discontinued? Is air ingress to storage containers credible?	1,23,24,49

(a) Included to indicate the consequences in the unlikely event that the inert cover gas degrades due to air ingress.

FIGURE 11. Comparison of Zircaloy Fracture Map Predictions with Data Points for Triple-Point Cracking and Dry Storage Conditions (based on concepts presented in Reference 16) (5- μm grain size)

The cladding failure mechanisms are related to stress and temperature in Figure 11 on a map of elastic modulus compensated stress (σ/E) versus the ratio of absolute melting temperature to absolute cladding temperature (T_m/T). Stresses (in MPa) are indexed along the right side of the figure, and temperatures ($^{\circ}\text{C}$) are scaled along the top. The slanted lines extending from the top to the bottom of the figure index estimated failure time isochrons (in seconds) for unirradiated Zircaloy at the values of stress and isothermal temperature traced by each isochron. The failure times are based on isothermal conditions and do not account for decay heat or temperature gradients. The map is subdivided into specific regions of failure behavior as described in the following sections. The space between the highest stresses (~ 100 MPa) anticipated for spent PWR fuel (see Appendix B) and the triple-point cracking zone in Figure 11 represents the estimated margin of integrity remaining between breached and nonbreached cladding at isothermal conditions. However, irradiation effects could decrease the triple-point cracking lower boundary and incipient defects in the cladding could raise the upper dry storage boundary and thereby reduce the estimated margin of integrity. Consequently, it cannot be assumed that breaches will not occur during dry storage.

Rapid Stress Rupture

Zircaloy cladding is susceptible to failure by rapid stress rupture when stresses exceed or approach the yield stress of ~ 500 MPa (Table 3). Such high stresses activate deformation mechanisms represented in Region A on the failure map in Figure 11. The required stresses are above the expected regime for dry storage (also shown in Figure 11). If large mechanical overloads were to occur, it is conceivable that ductile tearing, transgranular fractures would result (similar to those noted in high rate loss-of-coolant accident tests).⁽³⁴⁾ Damaging overload situations do not appear to occur, even during fuel handling or shipping; 30,000 or more spent fuel assemblies have been discharged and placed in storage. Cask shipments have involved over 5100 LWR assemblies without evidence of significant damage.⁽³⁰⁾

Failures by postulated thermal excursions occur at lower stresses and strain rates in Region A between 700°C and 900°C .⁽³⁴⁾ These abnormal events generate uniform cladding strains of $\sim 30\%$ and localized cladding strains

approaching 100%. Pinhole defects are generated in the high strain regions at low rates of temperature increase (Figure 12); the relationship between burst temperature and burst hoop stress is shown in Figure 13. Tests on unirradiated PWR whole rods at heating rates between 10^{-3} to 10^2 °C/s indicated that temperatures up to 765 to 800°C were required to cause fuel rod rupture. Rod internal pressures were 34 atm at room temperature.

Stress Corrosion Cracking

Both fission product attack and high stress are required for SCC; the joint action is regarded as the cause of pellet-cladding interaction (PCI) failures in-reactor on fuel that is subject to rapid changes in reactor power.^(11,15) Conditions for Zircaloy SCC coincide approximately with Region B in Figure 11.

In dry storage, fuel pellet temperatures are much lower than in-reactor (120 to 400°C versus 1200 to 1800°C); therefore, arrival of fresh fission products at cladding surfaces is correspondingly minimized. Fuel swelling ceases after reactor operation; therefore, local stresses generated by PCI should not exceed those resulting from reactor operation.

Assessments of fission product inventories and properties suggest that iodine, cesium, and cadmium are the most likely species to participate in PCI-type SCC.⁽¹¹⁾ A thermodynamic assessment suggests that released iodine reacts to form stable compounds that effectively remove iodine from further reaction.⁽⁴²⁾ Radiation may decompose iodine compounds,⁽⁴³⁾ but radiation levels in dry storage are orders of magnitudes lower than in the reactor ($<10^6$ R/h gamma versus $\sim 10^9$ R/h gamma in-reactor plus a high neutron flux). Laboratory studies have demonstrated that iodine, cesium, cadmium, and other species can promote SCC of Zircaloy over the temperature range of 100 to 400°C, but the required stresses (Region B) have been above the expected dry storage stress region shown in Figure 11. Laboratory tests conducted in the dry storage stress Region C (see Table 5 and Reference 3) did not produce SCC; active iodine levels were from 0 to 10^{-5} g/cm². The highest level was an order of magnitude above the threshold level for SCC in laboratory tests.⁽¹²⁾

FIGURE 12. Rod T3 After Testing at $\sim 40^{\circ}\text{C}/\text{h}$ Compared with Enlargement (3.2X) of Failure Site (Reference 34; scale in inches). Rod rupture occurred at $\sim 800^{\circ}\text{C}$, resulting in pinhole defects; rod internal pressure was 3.4 MPa at room temperature; prototype unirradiated PWR rod with UO_2 pellets.

FIGURE 13. Failure Temperatures and Stresses for Heating Rates from 3×10^{-3} to 30°C/s (based on information in References 34 and 50)

While the low incidence of rod failures over the wide range of dry storage test and demonstration conditions does not fully rule out SCC as a potential failure mechanism, it strongly suggests that SCC is not likely to be a major factor.

Strain Rate Embrittlement

Low-ductility failures in Zircaloy have been observed in tests conducted in the dry storage temperature range.^(40,41,46) The stresses for these tests spanned the boundaries for triple-point cracking (Region B in Figure 11). Because the long, wedge-shaped cracks caused grain boundaries to separate

uniformly from one cladding surface to the other, the corresponding failures occurred with generation of low strain. Cladding stresses during dry storage are not expected to exceed the 150-MPa hoop stress required to activate triple-point cracking; and, therefore, strain rate embrittlement is not expected to occur.

Slow Creep Rupture

Slow creep rupture occurs by the linking of small cavities that form along grain boundaries. This failure mode is responsible for the behavior in Region C on the failure map (Figure 11). Evaluations of Zircaloy stress rupture behavior during dry storage have suggested temperature limits below which the time for stress rupture in Region C exceeds the storage time. Blackburn et al.⁽¹⁷⁾ used the Larson-Miller parameter to predict temperature limits under isothermal conditions to prevent failure by stress rupture. Chin and Madsen⁽¹⁶⁾ used cumulative damage models to predict a temperature and time in storage limit to prevent cavity linkage from producing cladding failure. The resulting temperature limits are summarized in Table 8.

Oxidation of Zircaloy Cladding

The relationship between crud and oxide layers on Zircaloy-clad fuel is shown in Figure 14; zirconium oxides are generally tenacious on Zircaloy fuel rods. Typical EOL thicknesses are summarized in Table 9. Oxide thicknesses vary: along the length of a given rod, due to differences in temperature and radiation level; from rod-to-rod in a given reactor; and from reactor-to-reactor.^(39,51) In inert cover gases, reactive impurity inventories are not expected to be high enough to cause significant additional cladding oxidation. Even if air ingress should occur, corrective action seems feasible before oxidation would become significant.

In oxidizing cover gases, the decreasing temperature ramp tends to minimize cladding oxidation. Integration of the oxidation occurring from a starting temperature of 400°C, decreasing to 250°C over an ~2-year period results in only ~75 μm of oxide growth, which consumes only ~8% of the PWR cladding and only 6% of the BWR cladding.⁽¹⁾ Below 250°C, the oxidation rate is negligible over expected periods of interim storage (~40 years). Another limiting

TABLE 8. Models for Predicting LWR Spent Fuel Performance

Mechanism and Model	Benefits	Limitations	Predicted Temperature Limit, ^(a) °C	References
<u>Stress Rupture</u>				
• Larson-Miller parameter	Simplicity and broad experience in applications.	Data base limited to high stresses and unirradiated cladding that is atypical of spent fuel.	380	17
• Deformation and failure mapping	Accommodates changes in mechanisms and accounts for differences in metallurgical properties such as grain size.	Data base on irradiated Zircaloy limited to 140-h duration.	300 to 430 ^(b)	16
• Strain limit	Based on reaching ductility limit of Zircaloy cladding.	Requires accurate description of strain-time behavior.	400	3,4,10
<u>Stress Corrosion Cracking</u>				
• SCCIG model	Predicts crack initiation and growth as a function of metallurgical condition and environment.	Requires input on metallurgical condition and environment that is not now well defined.	150 to 400 ^(b)	15
• Crack growth model	Predicts incipient crack behavior.	Requires input on incipient cracks that is not now well defined.	150 to 400 ^(b)	12
<u>Fuel Oxidation^(c)</u>				
• Cumulative oxidation prior to power induction	Predicts storage time under variable temperature conditions.	Model not verified by variable temperature conditions. Data base on irradiated UO ₂ limited, but expanding.	150 to 180 ^(b)	49,52

(a) Temperature below which cladding failures are unlikely.

(b) Depends on fuel condition.

(c) Fuel oxidation is a consideration in the unlikely event of air ingress.

FIGURE 14. Relationship of Fuel Crud Layer to Fuel Cladding and Fuel Cladding Oxide (Reference 1)

TABLE 9. End-of-Life Oxide Thicknesses for Zircaloy-Clad Fuel

Reactor Type	Oxide Thickness, μm	
	Range	Average
BWR	10 to 125	25
PWR	3 to 60	20

situation arises in a sealed metal cask,^(a) where the oxygen inventory in the free cask volume generates only 3 μm of oxide on the entire fuel cladding surface or 60 μm if only 5% of the hottest cladding surface reacts.^(1,21) It therefore appears safe to conclude that oxidation is not a threat to the fuel cladding in scenarios currently regarded as credible.

(a) Basis for illustration: 24 PWR assemblies.

Hydride Reorientation

Hydrogen concentrations in spent Zircaloy cladding are normally in the range of 20 to 150 wppm. The temperature boundary for solubility of hydrogen in Zircaloy is shown in Figure 15. If the temperature falls below the solubility limit, zirconium hydrides precipitate. The hydrides normally orient in the circumferential and axial directions; however, if they form under a sufficiently high hoop stress, radial orientation can be induced. Zircaloy cladding ductility with radially oriented hydrides is reduced, even with hydrogen contents as low as 40 wppm.⁽⁴⁸⁾ During dry storage, the fuel at peak temperature conditions may be above the solubility limit (Figure 15). As it cools,

FIGURE 15. Effect of Spent Fuel Storage Temperature on Susceptibility of Zircaloy Cladding to Hydride Reorientation

it will reach the solubility limit. High cladding hoop stresses have caused hydrides to reorient during cooling; however, the stresses were intentionally maintained above stresses expected in cladding during dry storage.⁽⁵³⁾

Metallurgical examinations of irradiated Zircaloy-clad fuel nearly always show little tendency for preferred radial hydride orientation. Only if the hoop stresses were high enough to cause radial orientation and if the creep strain generated after hydride formation exceeded the ductility in the hydrided state could cladding breaches be produced. It is unlikely that the required creep strains would be generated after the fuel had cooled below the solubility limit; however, detailed analyses have not been performed over the full range of dry storage conditions.

Fuel Oxidation

UO₂ oxidizes in air or other oxidants. As the oxygen-to-uranium ratio reaches 2.67, low-density U₃O₈ forms and the cladding may crack due to fuel swelling. Since it cannot be assured that all fuel placed in dry storage will be free of cladding breaches, fuel oxidation is credible when spent fuel is stored in an oxidizing atmosphere at sufficiently high temperatures. The oxidation rate decreases with decreasing temperature; therefore, U₃O₈ formation and cladding cracking can be prevented if the temperature is sufficiently low or if the atmosphere is nonoxidizing. Data generated in tests with UO₂ pellets and defected fuel rods indicate that maximum storage temperatures below 180°C may be required to prevent oxidation to U₃O₈.⁽⁴⁹⁾ Tests are in progress to define the temperature below which oxidizing atmospheres will be tolerable. Until temperature limits are adequately defined, licensing of dry storage will probably be limited to nonoxidizing atmospheres.

Incipient Defects

As indicated by inspection of many metallographic cross sections, only a small fraction of irradiated Zircaloy rods develop incipient defects. Some shallow cracks are present when the cladding was fabricated.⁽⁴⁵⁾ Other incipient cracks develop during reactor operation, principally due to PCI effects. However, hydriding can also result in cladding cracks. Figure 16 shows an example of a ~40-μm deep incipient defect on the inner surface of an irradiated

FIGURE 16. As-Polished Incipient Crack on Internal Surface of Rod 120 of Shippingport Bundle 007 (Reference 54)

Zircaloy-clad rod from the Shippingport reactor. It would not be practical to inspect individual fuel rods by nondestructive methods to identify those with detectable incipient defects; nor is it assured that all incipient defects are detectable by current nondestructive testing technology.

Conditions for propagation of incipient defects in dry storage have been assessed and are not considered to offer a substantial threat to dry storage, due to a combination of low crack incidence⁽⁴⁾ and low probability for crack propagation in most dry storage regimes.⁽¹⁴⁾ For typical LWR spent fuel conditions, it has been predicted that maximum storage temperatures of $>400^{\circ}\text{C}$ will be allowable, with low expectation of SCC through-wall crack formation.⁽¹⁴⁾ Further, if a crack does propagate through-wall, the defect is expected to be small.⁽¹³⁾ While the effects of incipient defects in dry storage are not fully defined, their incidence and impacts are expected to be small. This conclusion is supported by dry storage experience, which indicates a zero failure rate on ~ 5600 rods exposed to inert cover gases. The one rod that failed in air in the FTT has not yet been inspected to determine the cause.

Contributions to Rod Internal Pressures from Actinide Decay in the Fuel During Interim Storage

Irradiated fuel contains an inventory of actinide species. The actinides undergo alpha decay, illustrated by the reaction:

Thus, helium will be generated inside fuel rods by decay of actinides. The isotopes that are the major contributors to helium accumulation inside fuel rods include ${}^{241}\text{Am}$, ${}^{238}\text{Pu}$, ${}^{242}\text{Cm}$, and ${}^{244}\text{Cm}$. Figure B.4 (Appendix B) analyzes helium pressures as a function of time and rod temperature. Inspection of Figure B.4 leads to the conclusion that actinide decay will make very small contributions to rod internal pressures over the time span of interim storage but can contribute significantly to rod internal pressures over time spans corresponding to repository storage.

3.3.3 Dry Storage Fuel Performance Models

Demonstrations and tests to verify LWR fuel behavior under dry storage conditions have been in progress for up to 8 years (Tables 5 and 6). Interim storage periods are expected to extend beyond the time periods for current demonstrations and tests, although at relatively low and decreasing temperatures after a few years. Modeling activities are useful to extrapolate fuel performance and to establish design and operating limits for extended operation of dry storage facilities. Models developed to predict fuel performance are described in Table 8; the strengths and limitations of each model are included. A common limitation in modeling is the relatively small data base for postirradiation stress rupture data for Zircaloy-clad fuel. Unirradiated data are generally used to assess Zircaloy behavior and are supplemented, where available, with data from irradiated cladding (see Figures 11 and B.3). Postirradiation creep data are also lacking, but a recent FRG study is addressing that subject.⁽²⁶⁾

3.3.4 Assembly Hardware Integrity

Assembly hardware, particularly lifting members, serve an important function during assembly emplacement and retrieval. Very few problems involving lifting members have occurred in thousands of fuel assembly movements. For fuel types that have had lifting problems, some surveillance and/or lifting tests may be required to assure that assemblies destined for dry storage have ample mechanical integrity.

3.3.5 Storage System Integrity

There is a background of operational experience for casks, dry wells, silos, and vaults⁽¹⁾ (see Appendix A). Storage casks have been licensed for shipment in the FRG; thus, they were subject to drop, fire, and impact tests. For storage, the FRG casks are fitted with a double-welded lid arrangement, which has provisions for monitoring to confirm integrity. Dry wells have functioned since 1964 in Idaho and since 1979 in Nevada. Silos have operated in Canada since 1975 and have been subject to integrity assessments. Vaults for dry storage have operated in Idaho since 1964.

4.0 TYPES AND IMPACTS OF CLADDING DEFECTS

This section discusses the types and impacts of cladding defects for Zircaloy-clad fuel. Fuel cladding defects from three sources will be examined:

- defects that form during reactor operation
- defects (if any) that form during normal dry storage conditions
- defects (if any) that form during abnormal dry storage conditions.

4.1 CLADDING DEFECT FORMATION DURING REACTOR OPERATION

A small fraction of the spent fuel inventory has cladding defects that developed during reactor service. The incidence of defective rods is estimated to be $<0.01\%$, which amounts to 4 or 5 rods per year in a 1000-MWe reactor.⁽⁵¹⁾ However, on an assembly basis, the fraction is higher because one failed rod in a BWR assembly is associated with either 47 or 62 intact rods; one failed rod in a PWR assembly is associated with ~208 to 272 intact rods. Certain early fuel batches and to a lesser extent some recent fuel batches have been subject to a higher incidence of fuel failures. Information on fuel failure mechanisms and histories is available in the literature.^(1,29)

Failed fuel can be stored without substantial impacts in spent fuel pools;⁽⁵⁵⁾ however, there is some concern about how to treat defective fuel in dry storage. One possibility--placing fuel in canisters (canning)--has been discussed.^(1,20) The disadvantages of canning include higher costs, some (probably small) personnel radiation exposure from canning and decanning operations, and handling and disposal or decontamination of the radioactive canisters. Unless compensating benefits can be identified, the canning approach seems marginal to unjustified. Exceptions include canning for rod consolidation or canning where required by a storage concept.

Cladding breaches range from minute pinholes or cracks a few microns wide (Figure 17) to large splits and holes up to several centimeters long (Figure 18). Cladding failures have been attributed to a variety of mechanisms, including overheating due to crud deposition (Figure 19), mechanical fretting, PCI (mechanical and chemical), cladding hydride formation (Figure 20), and

FIGURE 17. Narrow Crack in Zircaloy Cladding Due to Fission Product Attack

manufacturing defects. In-reactor fuel performance has significantly improved with understanding of the failure phenomena, which has led to improved fuel designs and reactor operating procedures.

4.2 CLADDING DEFECT FORMATION DURING NORMAL DRY STORAGE

Cladding defect development during dry storage appears to be a small but finite prospect. The development of a fuel rod defect in a dry storage test on a PWR assembly at the EMAD facility in Nevada was discussed earlier (Figures 8, 9, and 10). There is evidence that the primary cladding ruptures produce a very small defect that may be difficult to detect.⁽¹³⁾ Thus, the impact of a cladding defect that develops during dry storage is expected to consist of fission gas release to the sealed fuel container (cask or canister), with essentially no release of fuel particles. As with reactor-induced defects in dry storage, primary defects are not expected to propagate in inert cover gases.

4.3 IMPACTS OF CLADDING DEFECTS IN DRY STORAGE

The conditions expected during dry storage are compared with reactor service in Table 1. Dry storage conditions are less severe except for the

FIGURE 18. Defect That Developed in PWR Zircaloy Fuel Rod During Reactor Service. Comparison demonstrates that defect did not degrade in extended wet storage (Reference 56).

FIGURE 19. Zircaloy BWR Cladding Failure Due to Crud-Induced Overheating

FIGURE 20. Hydride Failure Initiating from Inner Cladding Surface

longevity of dry storage and the differential gas pressure across the cladding. Normal LWR fuel residence in-reactor is usually only 3 or 4 years, but interim dry storage may extend for 20 to 40 years. The total dry storage time prior to repository disposal may be greater than 50 years (for example, the MRS concept). The high-pressure reactor environment that supports the cladding exterior and suppresses expansive creep deformation during reactor service is not present during dry storage. Thus, resulting cladding hoop stresses must be assessed in terms of potential impacts on cladding behavior in storage. Figure 11 provides a perspective regarding the relationship of cladding hoop stresses for fuel in dry storage to expected cladding failure regimes (see discussion in Section 3.3.2).

Cover gas selection is a key consideration. In inert gases, there is no mechanism for further propagation of cladding defects. Because the internal gas pressure was released while the fuel was in the reactor, further rupture from hoop stress is precluded. Inert gas has only small amounts of reactive impurities; therefore, reactions with the cladding or exposed fuel will not be significant. Only if substantial water ingress into the rod occurred during reactor residence (water logging) would a reactive gas be available. Recent studies of defective BWR rods suggested that the moisture inventories inside the rods amounted to only a few milliliters.⁽⁸⁾ Experience to date suggests that most of the moisture would be removed during cask drying operations.

When a spent fuel rod with large defects was heated in argon at 325°C for 2100 h, further degradation was not discernible (see Figures 21 and 22). In contrast, a companion rod with large defects underwent gross degradation (Figure 23) when exposed to replenished air at 325°C.⁽⁸⁾ There is other evidence that air reacts with exposed UO₂ pellets to form U₃O₈, resulting in a volume increase that causes cladding defects to enlarge.^(8,22-24) Studies are under way in several countries to define a thermal regime that is satisfactory for air storage of UO₂ fuel. Until that regime is defined, inert gas storage appears to offer an approach that will not further degrade fuel with defects.

A very small fraction of breached fuel rods have defects that are large enough that pellets or fuel fragments might emerge during fuel handling operations. Reasonable measures to exclude larger cladding defects from dry storage seem to be justified. However, pellets are generally tightly held by the cladding even when exposed at a cladding defect.

4.4 CLADDING DEFECT FORMATION DURING ABNORMAL EVENTS

The range of abnormal events that needs to be analyzed for dry storage has not been fully defined because some events are site specific. The most likely abnormal events would be:

- a fire, although it probably would not be of major proportions for storage-only modes
- a change in cask or silo orientation (for example, vertical to horizontal)
- partial or complete burial of the storage system by debris, etc.

The thermal inertia of multi-assembly storage systems precludes rapid approaches to overtemperature conditions; therefore, abnormal events would tend to slowly raise the cladding temperature. However, it does not seem credible that cladding temperatures would approach the high temperatures (765 to 800°C) required in slow temperature ramps to produce cladding rupture (Figure 13 and Reference 34).

Neg. A377

Neg. A383

a) Before Testing in Argon

Neg. A520

Neg. A519

b) After Testing in Argon

FIGURE 21. Reactor-Induced Breaches in BWR Spent Fuel Rod Before and After Testing in Argon for 2100 h at 325°C

4.5 IMPACTS FROM MOBILE RADIOACTIVE SPECIES

Three types of mobile radioactivity may be generated inside a container during dry storage: gaseous species, principally krypton and tritium; dislodged crud particles; and fuel species. The gaseous species would arise in significant amounts if cladding failures were to develop during dry storage. The inventories of mobile radioactive gases depend on fuel type (BWR versus

Neg. A389

Neg. A518

a) Before Testing in Argon

b) After Testing in Argon

FIGURE 22. Reactor-Induced Hydride Breach in BWR Spent Fuel Rod Before and After Testing in Argon for 2100 h at 325°C. Testing did not result in detectable degradation.

PWR), fuel burnup, and fuel operating conditions in-reactor. An assessment of ^{85}Kr impacts if released during dry storage concluded that up to 7% of the rods could fail annually and release mobile gases to the atmosphere without exceeding ALARA guidelines.⁽⁵⁷⁾ The total number of rods in storage was assumed to be 15,840.

To date, fuel inspections conducted after periods of dry storage have not shown large amounts of particulate crud. As with the mobile gases, crud particles will be contained by the sealed dry storage vessel (cask or canister).

Release of fuel particles from fuel rods stored in inert cover gases is highly unlikely. Defects that form during dry storage are expected to be small,⁽¹³⁾ limiting escape of fuel particles. Very few LWR rods have reactor-induced defects with configurations that could release large fuel particles.

The impacts of mobile radioactivity inside the dry storage container during fuel retrieval operations appears to be the most significant consideration relative to the species considered above. NRC/HEDL/EG&G tests indicate that there is some fission product migration from a breach in a spent fuel rod.⁽²⁴⁾

Neg. A365

3X

Neg. A622

1X

a) Before Testing in Air

b) After Testing in Air

FIGURE 23. Reactor-Induced Breach in BWR Spent Fuel Rod Before and After Testing in Air for 2100 h at 325°C

When fuel is retrieved from the dry storage vessel using wet operations, the pool water will contain the particulate species, which can then be recovered by the pool cleanup system. Soluble species, such as cesium, will be removed by the ion-exchange system. If large gaseous inventories are anticipated, they can be channeled by poolside hoods and collected for storage or controlled release; however, it has been estimated that free ^{85}Kr inventories will not exceed site boundary limitations, even for numerous rod failures as indicated earlier.⁽⁵⁷⁾

It is not clear that LWR fuel recovery from interim storage under dry conditions will be utilized on a large scale in the United States. Dry handling of contaminated assemblies is routinely accomplished in hot cells. If other dry recovery procedures are foreseen, then procedures to deal with mobile radioactive species should be defined in advance.

4.6 DEFECT DETECTION

Irradiated fuel batches can be screened for cladding defects in several ways:

- Reactor off-gas levels provide an indication of fuel soundness.
- Frequently, fuel batches with suspected cladding defects (leakers) are inspected in the reactor pool during reactor outages, using

nondestructive methods: visual, sipping, ultrasonics, eddy current, profilometry, and gamma scanning. Fuel inspection records may include results of hot cell examinations of one or more fuel assembly from a given batch, providing insights to fuel condition.

- Inspection of previously uninspected fuel for cladding defects after extended wet storage has a relatively limited experience base. Sipping relies on detection of isotopes escaping from a cladding defect. Therefore, radioactive decay and leaching by pool water reduce the effectiveness of sipping with time after discharge. However, nondestructive detection methods are available that do not degrade with storage time.⁽⁵⁸⁾

The following considerations may be useful in deciding what level of cladding defect assessment to apply to a given fuel selection process for dry storage:

- As indicated in Table 2, systematic studies in several countries and by the IAEA⁽²⁹⁾ suggest that cladding defects have not developed in Zircaloy-clad fuel during wet storage. Similarly, there is evidence from FRG, U.S., and United Kingdom studies suggesting that reactor-induced Zircaloy cladding defects do not propagate during wet storage.
- There are secondary in-reactor mechanisms that cause primary defects to enlarge. In inert dry storage cover gases, these secondary mechanisms are absent. As indicated earlier, this observation was demonstrated experimentally on a BWR Zircaloy-clad rod with reactor-induced defects, exposed to an argon cover gas at 325°C. In a 2100-h test, the cladding defects did not change configuration (Figures 21 and 22).
- A reasonable approach regarding defective rods in dry storage is to exclude rods with large defects in a configuration that could result in loss of fuel fragments during assembly insertion or retrieval. The spent fuel inventory appears to include very few defects of this type.

5.0 SUMMARY AND CONCLUSIONS

Analysis of dry storage tests and demonstrations, dry storage modeling efforts, and Zircaloy technology leads to the conclusion that irradiated Zircaloy-clad LWR fuel is durable over a broad range of dry storage conditions in inert cover gases. Approximately 5600 Zircaloy-clad rods have been monitored in dry storage tests and demonstrations over the temperature range from 50 to 570°C. The positive results now available provide ample evidence to justify licensing Zircaloy-clad fuel for extended dry storage in inert cover gases at cladding temperatures up to 380°C. This temperature represents the lower limit of the range for licensed demonstrations conducted in the FRG. Other FRG tests and demonstrations are under way with maximum temperatures from 400 to 450°C. The 380°C temperature represents a conservative position that should be re-evaluated and possibly increased after the results from higher temperature tests and demonstrations have been analyzed.

Zircaloy-clad fuel in tests and demonstrations (Tables 5 and 6) has been tested in several thermal regimes:

- normal maximum temperatures (380 to 430°C) - Involves a relatively small fraction of the monitored rods; both BWR and PWR rods and assemblies are represented; maximum residence for high-temperature rods on a prototypic temperature ramp is ~1.5 years (385°C down to 250°C for the hottest KWW rods). The temperatures of the coolest KWW rods range from 180 to 120°C. The hottest KKS rods began at 430°C and had decreased to 340°C at the last integrity surveillance.
- intermediate temperatures (200 to 380°C) - Involves the major fraction of monitored rods; demonstrations that include the hottest rods (KWW and KKS) have lower temperature rods that began in or have passed through this regime as indicated above; assemblies that began in this regime have been in dry storage for up to 5 years at NTS.
- lower temperatures (<200°C) - Involves BWR, PWR, PHWR, and WR-1 rod types, including monitored rods that have been in dry storage for up to 5 years.

- abnormal regimes ($>450^{\circ}\text{C}$) - Includes rods tested at 480 to 570°C for up to 325 days without rod failures; unirradiated rods tested at 765 to 800°C ruptured in 1 to 2 days; the ruptures resulted in pinhole defects.

Internal rod pressures are not yet defined for all rods in the tests and demonstrations. EOL pressures for Turkey Point rods range from 300 to 400 psi (RT); FRG assemblies in demonstrations were discharged since 1980 and therefore represent relatively recent fuel designs and characteristics. Burnups up to 28,000 MWd/tU (BWR) and 43,700 MWd/tU (PWR) are included in the demonstrations. Therefore, EOL pressures are expected to include or exceed both BWR and PWR ranges in U.S. assemblies that would be candidates for dry storage. In fact, older U.S. assemblies (fabricated prior to ~ 1970) would almost certainly have substantially lower EOL pressures than those included in FRG and U.S. tests and demonstrations, due to lower burnups and fill gas pressures (see Figures B.1 and B.2 in Appendix B).

Maximum cladding temperatures for the hottest fuel stored in metal casks may be $\sim 60^{\circ}\text{C}$ after 40 years (see Figure 6). Canadian rods passing through that range have been examined after ~ 4 years of dry storage without evidence of cladding degradation. The rods had relatively low burnups typical of PHWR fuel. The coolest rods in the KWW test will be near 60°C during the 2-year demonstration. Figures B.1 and B.2 indicate that the internal rod pressures decline as the fuel rods cool. Estimates of rod pressure based on maximum rod temperatures will be conservative due to temperature gradients along the rods and across the storage facility (see Figure 5).

Dry storage tests and demonstrations have been conducted principally in inert cover gases (argon and helium). No intact rods have failed, and defects in a BWR rod did not propagate in 88 days at 325°C . Two KWO demonstrations were conducted in moist nitrogen, where radiolytic formation of more aggressive species such as HNO_3 was a prospect (see Figure 8). Over the conditions of the two demonstrations (300°C , declining for 60 days; 400°C , declining for ~ 1 year), the environment did not cause detectable degradation of the assemblies, based on the surveillance methods indicated in Table 6.

A few test rods and one full assembly have been exposed to air. Rods with defective cladding were clearly susceptible to continuing degradation in air at 230°C⁽²⁴⁾ and 325°C,⁽⁸⁾ due to oxidation of exposed UO₂. Several tests are in progress to define a temperature where UO₂ oxidation will not cause significant fuel degradation.

Alpha decay of actinides generates helium that contributes to pressures inside the rods. Figure B.4 indicates the expected magnitudes of the pressure increase for two temperatures over a wide time span. For the time spans expected for interim storage (40 years), the pressure increases would be much less than 1 MPa at room temperature.

While current experience suggests that the incidence of cladding failures will be very low under dry storage conditions, some rod failures cannot be ruled out, as evidenced by the failure of one rod in a dry storage test at EMAD at ~270°C. However, any defects that develop during dry storage are expected to be very small. Methods are available to detect most assemblies having rods with reactor-induced defects, although elimination of every defective rod cannot be assured. However, both theory and experience indicate that there is no mechanism that will cause cladding defects to grow during storage in inert cover gases.

The potential for mobile radioactive species inside the storage chamber arises from three sources:

- crud (radioactive oxide deposits) that may spall from exterior rod surfaces during dry storage
- radioactive gases expelled if a cladding defect develops during dry storage (principally ⁸⁵Kr and tritium for older fuel)
- fuel particles and fission products if permitted by defect geometries and fuel characteristics.

Some degree of crud release from rod surfaces is likely, although experience to date suggests that amounts may be minimal. Radioactive gases would be retained inside the storage chamber until the fuel is retrieved. However, analyses suggest that free ⁸⁵Kr from numerous rods (7% per year) would not exceed guidelines for offsite releases, even if the gases escaped.⁽⁵⁷⁾ The

very low incidence of large reactor-induced defects and the absence of progressive fuel degradation in inert gases suggests that fuel particles and fission products are not likely to develop substantial mobile inventories.

For sealed storage chambers, impacts of mobile radioactive species would arise only when fuel is retrieved for reprocessing or repackaging, if that is necessary, for final disposal. Methods to manage mobile radioactive species need to be defined in advance and need to be consistent with specific retrieval operations.

To expand dry storage options available to utilities, further licensing considerations should include an acceptable thermal regime for storage in air and storage of consolidated fuel. Consolidation is now the base case for repository disposal of spent fuel,⁽⁵⁹⁾ providing an additional consideration for interim storage of consolidated fuel. To the extent that downstream operations and criteria are defined, interim storage operations should simplify, where practicable, subsequent operations.

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APPENDIX A

DRY STORAGE OPERATIONS

TABLE A.1. Installations and Demonstration Tests for Dry Storage of Irradiated Nuclear Fuel^(a)

Type of Storage	Location	Available Storage Spaces	Assigned Storage Spaces	Fuel Type/Contents	Cladding Type	Heat Generation per Unit, kW	Operational History																				
							64	65	66	67	68	69	70	71	72	73	74	75	76	77	78	79	80	81	82	83	
Dry well	Washington-Hanford (basalt)	2 1	2 1	Electric Electric	-- --	2, 5, 9 2																					
	Nevada Test Site- Climax (granite)	17	11	PWR	Zircaloy-4	2.4																					
	EMAD (soil)	4	3	PWR	Zircaloy-4	0.9																					
	EMAD (soil)	1	0	Electric	--	3, 1, 2, 3																					
	Washington-Hanford (soil)	1	0	Electric	--	1.0																					
	Idaho-INEL (soil)	61	14	LMFBR-Fermi 1	SS	0.7																					
			47	HTGR-PBI, Core 1	C	0.8																					
		361	99	LMFBR-EBR-II	SS	0.0																					
			621	Miscellaneous radioactive scrap	--	0.0																					
		Kansas-PSV (salt)	15	14 1	ETR/electric heater Electric	Al --	1.5 1.5																				
Vault(b)	Nevada Test Site- EMAD	24	13	PWR	Zircaloy-4	2.0																					
	Wylfa, Wales	28,800	NA(c)	AGR-Magnox	Mg-Al-Be(d)	0.06																					
	Idaho-INEL	144	--	LMFBR-EBR-II	SS	--																					
		40	--	LMFBR-EBR-II	SS	--																					
		636	445	HTGR-FSV	SiC	--																					
			67	HTGR-PBI, Core 2	C	0.5																					
			124	HTGR-Rover	C	0.0																					
Silo	Nevada Test Site- EMAD	2	1	PWR	Zircaloy-4	0.9																					
	Washington-Hanford	1	1	Electric	--	5.0, 10.0																					
	Canada-WNRE	7	1	WR-1	Zr-2.5 Nb	2.0																					
			1	WR-1	Zr-2.5 Nb	2.0																					
			1	HPWR-CANDU	Zircaloy-4	2.0																					
			1	Electric	--	2.0, 11.0																					
			1	WR-1	Zr-2.5 Nb	2.0																					
			1	Electric	--	2.0, 5.0																					
Metal cask	FRG-Wuerzgassen	1 cask	16/cask	BWR	Zircaloy-2	1																					
	FRG-Stade (Karlsruhe)	1 cask	4/cask	PWR	Zircaloy-4	5																					
	FRG-Julich	1 cask	NA	GCR	Ceramic	--																					
	Switzerland-Wuerenlingen	1 cask	NA	Test reactor	Zircaloy-2	--																					

(a) Adapted from Wright, J. B. 1981. "Spent Fuel Dry Storage--A Look at the Past, Present, and Future." Presented at the Fuel Cycle Conference '81, March 15-18, 1981, Los Angeles, California. See Glossary for definition of terms.

(b) Recent vaults in France, FRG, and Japan not included.

(c) NA = not available.

(d) Magnesium alloy cladding for gas reactor fuel; typical composition = Mg-0.9 Al-0.0025 Be.

APPENDIX B

NONPROPRIETARY LWR FUEL MODELING CODES

APPENDIX B

NONPROPRIETARY LWR FUEL MODELING CODES

Hoop stress is a key parameter used in rapid stress rupture, creep deformation, creep rupture, and stress corrosion cracking analyses. It is computed from the fuel rod internal gas pressure (P) and cladding dimensions, using the following formula for thin-walled tubing:

$$\text{Hoop Stress} = \frac{PD_m}{2t} \quad (\text{B.1})$$

where D_m is the midwall tube diameter and t is the wall thickness. The internal gas pressure results from prepressurization during fuel rod manufacture and fission gas release from fuel pellets during irradiation. Fuel modeling codes developed and based on end-of-life (EOL) fuel rod gas release data are used to compute the fuel rod pressure. Measurement of fuel rod EOL pressure during reactor operation requires pneumatic connection of instruments to the fuel rod; this method is therefore used only for developmental and verification purposes on selected fuel rods. EOL pressures can also be determined by puncturing rods and collecting and measuring released gases. The free volume inside the rod is also measured to define the pressure.

Selected nonproprietary fuel codes used to compute fuel rod pressures are presented in Table B.1. These codes take into account the fuel design, power history, and burnup as well as volume changes due to thermal expansion, fuel swelling, and cladding creep. Some codes also compute cladding stress with allowance for cladding wastage. Some typical EOL pressures were computed for typical pressurized water reactor (PWR)^(B.7,B.8) and boiling water reactor (BWR) fuel designs as a function of temperature using assumed gas release and measured void volumes in the GAPCON-THERMAL-2 fuel code.^(B.9) The results are plotted in Figures B.1 and B.2. The computed pressures were converted into hoop stresses using Equation (B.1) and placed on failure maps for the nominal PWR burnup case (36,000 MWd/MTU) in text Figure 11 and for the high-burnup case (50,000 MWd/MTU) in Figure B.3. Comparison of Figure 11 and Figure B.3 reveals

TABLE B.1. Fission Gas Release Predicted by Nonproprietary LWR Fuel Codes

<u>Fuel Code</u>	<u>Comparison of Prediction to Fission Gas Release Data Base^(a)</u>	<u>Reference</u>
GAPCON-THERMAL-3	Underpredicts at low fuel temperatures	B.1
FRAPCON-2	Underpredicts high release	B.2
COMETHE	Underpredicts	B.3
FCODE	Underpredicts	B.3
GCODE	Overpredicts	B.4
PATGA	Underpredicts	B.4
PATGB	Underpredicts	B.4
TRAFFIC	Overpredicts	B.5
CARO	Overpredicts	B.6
GAPCON-THERMAL-2	Overpredicts	B.9

(a) The fuel codes overpredict some data points and underpredict others; the statistical bias with a specific data set is shown here.

that special consideration for storage of high-burnup fuel may be necessary since the stresses approach Region B for failure by triple-point cracking. The pressures in BWR spent fuel rods are too low to be of concern for stress rupture below 400°C as shown in Figure B.2.

Helium is generated by the emission of alpha particles during radioactive decay of certain actinide isotopes during long-term storage. Computations made to determine the pressure buildup in fuel rods over extended storage or disposal are illustrated in Figure B.4 for a PWR fuel rod with a burnup of 36,000 MWd/MTU for several storage temperatures. Additional pressure increases due to actinide decay are less than 1 MPa during interim storage and would have little impact on fuel cladding stresses for the duration of interim and

FIGURE B.1. Plenum Gas Pressure for PWR Spent Fuel Computed from Measured EOL Void Volumes and Indicated Fission Gas Release Assumptions (pressure/temperature relationship from GAPCON-2)

FIGURE B.2. Plenum Gas Pressure for BWR Spent Fuel Computed from Measured EOL Void Volumes and Indicated Fission Gas Release Assumptions (pressure/temperature relationship from GAPCON-2)

FIGURE B.3. Comparison of Zircaloy Fracture Map Predictions with Data Field for Failure by Triple-Point Cracking and Dry Storage Conditions for PWR Fuel Rods with 50,000-MWd/MTU Burnup (5- μ m grain size)

FIGURE B.4. Pressure Increase Due to Helium from Actinide Decay

monitored retrievable storage. In very long-term storage (1,000 to 10,000 years), the pressure contribution from actinide can become significant. This pressure is illustrated in Figure B.4.

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APPENDIX C

HISTORIES OF TURKEY POINT ASSEMBLIES AT THE NEVADA TEST SITE

APPENDIX C

HISTORIES OF TURKEY POINT ASSEMBLIES AT THE NEVADA TEST SITE

Seventeen Turkey Point pressurized water reactor assemblies were shipped to the EMAD facility at the Nevada Test Site in 1978 and 1979. One other assembly was examined at a hot cell at the Battelle Columbus Laboratories (BCL) using nondestructive and destructive techniques. That assembly remains at BCL where rods have been used in whole-rod furnace tests (see text Table 5). Five of the Turkey Point assemblies now at EMAD were sent to BCL for nondestructive evaluation of selected rods, and the rods were reinserted before the assemblies were sent to EMAD. The 17 assemblies have been placed in dry storage demonstrations (see text Table 6) in deep and surface dry wells, a silo, and an air-cooled vault (referred to as the lag storage pit in Figure C.1).

The Turkey Point assemblies have been subject to several operations, including calorimetry, placement inside welded stainless steel canisters, and gas sampling, as indicated in Figure C.1. Two assemblies were placed in the fuel temperature test (FTT) to calibrate cladding temperatures against canister temperatures. Assembly B02 was removed from the silo, visually inspected, and then placed in the FTT in an air cover gas. All the other assemblies have been exposed in helium, except for the other two earlier FTT assemblies, which were briefly stored in vacuum and air environments.

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FIGURE C.1. Dry Storage Demonstrations of Turkey Point Assemblies (PWR) at the Nevada Test Site (Reference C.1)

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