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FOREIGN EXPERIENCE ON EFFECTS OF EXTENDED DRY STORAGE ON THE INTEGRITY OF SPENT NUCLEAR FUEL

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ABSTRACT

This report summarizes the results of a survey of foreign experience in dry storage of spent fuel from nuclear power reactors that was carried out for the U.S. Department of Energy's (DOE) Office of Civilian Radioactive Waste Management (OCRWM). The report reviews the mechanisms for degradation of spent fuel cladding and fuel materials in dry storage, identifies the status and plans of world-wide experience and applications, and documents the available information on the expected long-term integrity of the dry-stored spent fuel from actual foreign experience. Countries covered in this survey are: Argentina, Canada, Federal Republic of Germany (before reunification with the former East Germany), former German Democratic Republic (former East Germany), France, India, Italy, Japan, South Korea, Spain, Switzerland, United Kingdom, and the former USSR (most of these former Republics are now in the Commonwealth of Independent States [CIS]). Industrial dry storage of Magnox fuels started in 1972 in the United Kingdom; Canada began industrial dry storage of CANDU fuels in 1980. The technology for safe storage is generally considered to be developed for time periods of 30 to 100 years for LWR fuel in inert gas and for some fuels in oxidizing gases at low temperatures. Because it will probably be decades before countries will have a repository for spent fuels and high-level wastes, the plans for expanded use of dry storage have increased significantly in recent years and are expected to continue to increase in the near future.

CONTENTS

TABLES

1.0 INTRODUCTION

Spent fuel from civilian nuclear power reactors in the United States will be disposed of in one or more deep geological repositories. The U.S. Department of Energy's (DOE) Office of Civilian Radioactive Waste Management (OCRWM), which is responsible for disposal of the civilian spent fuel, plans to start accepting some of the spent fuel from the civilian nuclear power stations in 1998 for interim storage in preparation for disposal. Disposal is planned to start in the year 2010 at the Yucca Mountain, Nevada, site if it is found to be suitable, and if the repository has been developed to accept spent fuel by that time.

A significant amount of spent fuel is planned to be placed in dry storage in the U.S. (some for 20 or more years) until the spent fuel is disposed in the final repository. Some of this storage will be at reactors and some is planned to be at a federal monitored retrievable storage (MRS) facility. The extended dry interim storage will be needed because:

- DOE/OCRWM will not be able to accept spent fuel until 1998 (for interim dry storage) at the earliest
- the repository will not be available for disposal of spent fuel until 2010 at the earliest
- some U.S. nuclear power utilities have already implemented some dry storage and many more will need dry storage during the next 20 years, because their in-pool storage capacity is or will be used up.

Although dry storage of spent nuclear fuel is now a recognized and licensed technology in the U.S. and in some other countries, it is important to know the results of actual dry storage experience on the integrity of spent fuel. This information is important because after the extended dry storage period, the U.S. spent fuel must be handled while loading it into transportation casks at the nuclear power stations, while unloading it from transportation casks at OCRWM's repository or MRS facility, and while loading it into disposal packages at OCRWM's repository. This subsequent handling must be done safely and effectively. In addition, the condition of the spent fuel after encapsulation in the final disposal package could affect its safety in the repository.

Because of these factors, OCRWM asked the International Program Support Office (IPSO) at Pacific Northwest Laboratory (PNL)[a] to carry out a survey of foreign experience in dry storage of spent fuel from nuclear power reactors and to identify any known impacts on the integrity of the dry-stored spent fuel. The results of this survey are documented in this report. Information was sought about countries known to have some experience with dry storage of spent nuclear fuels from the PNL/IPSO resource center, additional literature searches, and from some personal contacts with domestic and foreign staff.

Except in the Summary and Background sections, information on countries with current or past significant dry storage activities is presented in separate sections by country (including references), using the same outline for each. These countries are Canada, the former West Germany, France, India, Japan, Switzerland, and the United Kingdom. The dry storage status in countries with limited activities (i.e., Argentina, the former East Germany, Italy, South Korea, Spain, and the former USSR) is discussed briefly in Section 11. Countries with no known activities in dry spent fuel storage are not discussed. However, at the time of publication of this report, considerations of dry spent fuel storage have been indicated by several countries in Eastern Europe. Because of the current uncertainties of these future activities, they are not covered in this report.

Areas where no information could be found are identified. To provide background on dry spent fuel storage for each country, information is first presented on the country's overall nuclear power situation. Then, information on its nuclear power reactors, its spent fuel management policy, and the overall status of its spent fuel storage program is presented. Information from

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the literature cited is paraphrased, with a minimum of interpretation by the authors. When interpretations by the authors are included, they are indicated by enclosing the information in brackets.

2.0 SUMMARY AND CONCLUSIONS

A survey of foreign experience in dry storage of spent fuel from nuclear power reactors was carried out for OCRWM and is presented in the subsequent sections of this report. The objectives were to identify the status of worldwide applications, plans, and experience, and to document available information on the expected long-term integrity of the dry-stored spent fuel based on actual experience. This study was a general survey of pertinent foreign activities; determining the validity of the subsequent results and conclusions from the countries surveyed was beyond the scope of this study.

2.1 SUMMARY

Highlights of the information obtained about each of the countries surveyed relative to the objectives of this study are summarized in Table 2.1. Thirteen foreign countries were reviewed that were known to have some activities in dry storage of spent fuel. The 13 countries identified in Table 2.1 are arranged in two rather arbitrary categories: the first (Part A) includes those seven countries believed to have somewhat active programs in dry storage; and the second category (Part B) contains those countries with less active programs or whose activities are planned for the future. The dry storage activities in countries listed in Part A were surveyed in more detail than those in Part B. Potential significant activities for dry storage of spent fuel in some other countries (particularly in eastern Europe) that have developed at the time of publishing this report are not included.

The seven countries in Part A of Table 2.1 all have at least a few metric tons of uranium (MTU) or heavy metal (HM) in spent nuclear fuel in dry storage in vaults or in casks. However, only Canada and the United Kingdom (U.K.) have implemented dry storage on an industrial scale. By the end of 1991, Canada had about 600 MTU of its CANDU fuels stored in dry concrete casks at several sites. The Canadian fuel (Zircaloy-clad natural UO_2) is stored in air at cladding temperatures below 150°C. The U.K. has stored hundreds of MTU of spent fuel from its Magnox reactors at a central vault storage facility at one nuclear power station. The U.K. stores the Magnox fuels (uranium metal

TABLE 2.1. Summary of Dry Spent Fuel Storage Experience in Selected Countries

Country	Current Dry Storage	Future Dry Storage	Research & Development	Longevity of Dry Storage Experience	Expected Duration of Fuel Integrity at Clad Temperature
Part A: Dry Storage Experience in Countries with Active Programs					
Canada	600 MTU of CANDU fuel (natural UO_2 clad in Zircaloy) in concrete casks in air in 1991; T is <100°C to 150°C.	Hundreds of MTU/yr to be loaded into casks in next few years. Technology is licensed. AECL is planning on using ventilated vaults, CANSTOR.	Since 1974, and to continue past 2000. Exam. of clad and fuel degradation in air, 100-150°C. Heat transfer work. Developing concrete casks for storage and transportation.	Test storage since 1980. Exams. after 8.3 yr to date.	General: 50 to 100 yr Cladding: 100 to 1000 yr at 100°C in air Defected: >8 yr at 150°C in dry air and in moist, limited air.
Federal Republic of Germany	0.62 MTHM of HTGR fuel (from HTGR/AVR) in cans in hot cells in He at 40°C-170°C. 6.9 MTHM in HTGR/THTR fuel (UO_2 + ThO_2 spheres) in cans in A-R vault in He at <320°C.	Two 1500-MTU AFRs for Castor and TN casks are built for LWR fuel (and pebble-bed fuel and vitrified HLW) but not yet operating. Technology is licensed.	From 1979 to 1986, tested 3000 LWR rods. Tested durability of cladding in inert gases at 250-430°C and did heat transfer work. Developing cask for stg./transp./disp.	Tests on LWR rods lasted up to 2 yr.	General: Up to ~100 yr for LWR fuels Cladding: Up to ~100 yr at initial max. temp. of 420°C in inert gas for LWR fuels Defected: TBD.
France	A concrete vault for 9.5 MTHM for FBR fuel (UO_2 + PuO_2 in ss. clad) at <640°C. A second concrete vault started up in 1990 for experimental fuels using He-filled canisters at <180°C.	Continue storage for up to 200 MTU experimental fuels for decades, with possible future expansion. Vault design available for LWR fuels at La Hague if needed.	None found.	Experimental fuels storage started in 1990. FBR fuel stored since 1985.	General: ~50 yr for experimental fuels in inert gas Cladding: No information found Defected: No information found.
India	Four indigenous storage/transportation casks store 20.6 MTU of fuel from BWRs in air since 1987. Temp. unknown.	No information on any more planned; vault planned for vitrified HLW.	Heat transfer work.	Casks loaded in 1987.	No information found.
Japan	15 MTU of research reactor fuel (natural U clad in Al) in cans in He in "dry-well" in vault since 1982. No temp. information found.	Space for 15 MTU more research fuel.	Oxidation tests in air and Ar on Zircaloy and UO_2. Cladding creep tests and designs of cast iron casks. Durability tests of metallic seals.	Started dry storage of research reactor fuel in 1982. Exams after 5 yr.	General: 30 yr or more for LWR fuels Cladding: At least 30 yr for LWR fuels in air at 328°C; in inert gas at 350 to 430°C Defected: About 30 yr for LWR fuels in air at 160°C.
Switzerland	2.5 MTU of research reactor fuel (U clad in Zircaloy) in Castor cask in 1983 in He at 180°C.	AFR planned for up to 1550 MTU of fuel and/or HLW in Castor casks. Technology is licensed.	None found.	Research reactor fuel loaded into cask in 1983. No exams. of fuel.	No information found. Apparently use FRG data.
United Kingdom	Hundreds of MTU of Magnox fuel (U in Mg-Al alloy) in one vault in CO_2, followed by storage in two other vaults in air. Initial storage temp. is up to 365°C in CO_2 and 150°C in air.	Continued use of existing vaults for Magnox fuel. Vault planned by Scottish Nuclear for ~2000 MTU of AGR fuels. Technology is licensed.	Design/development of vault storage for Magnox and AGR fuels. Fuel and cladding oxidation tests in air at 250 to 450°C. In early stages of R&D for AGR fuels.	Storage started in CO_2 in 1972 and in air in 1979. Fuel is usually stored 1 to 4 yr before reprocessing. Some exams. after 4-5 yr.	General: Few years to several decades for Magnox fuel in CO_2 then air under U.K. vault conditions. Cladding: No information found. Defected: About 30 yr for AGR fuels in air at 100-150°C.

TABLE 2.1. (contd)

Country	Current Dry Storage	Future Dry Storage	Research & Development	Longevity of Dry Storage Experience	Expected Duration of Fuel Integrity at Clad Temperature
Part B: Dry Storage Experience in Other Countries					
Argentina	No storage to date.	Plan to start use of Canadian concrete casks for CANDU fuel in 1993. Plan for 5 to 33 yr of fuel production.	Started development of dry well, but status is unknown. Did heat transfer studies.	No experience to date.	No information found. Apparently using Canadian data.
Former German Democratic Republic	No storage to date.	No plans were identified before unification with FRG. Recent indications are for an AFR holding 300 to 700 MTU using dry storage casks.	Heat transfer studies for storage casks.	No experience to date.	No information found.
Italy	Fuel from Magnox reactor stored 15 to 20 yr in air and nitrogen in sealed cans in a pool. ~2 MTU from specialty fuels stored in a hot cell.	Dry storage of Magnox fuel was discontinued when fuel shipped for reprocessing. Dry storage of specialty fuel to continue. No other plans.	None found.	Magnox fuel stored 15 to 20 yr. Recent exam of Magnox fuel showed much cladding corrosion.	Much cladding failure after 15-20 yr, but fuel mostly intact. Failures probably due to high-temp. storage in oxidizing gas.
South Korea	No storage to date.	Canadian-type concrete casks ordered for 700 MTU of CANDU fuel, starting 1997.	Starting generic dry storage R&D. Heat transfer studies for vault storage.	No experience to date.	No information found. Apparently using Canadian data.
Spain	No storage to date.	Plan for >500 MTU storage in NAC transport/storage casks for LWR fuel in inert atmosphere. Central AFR planned for ~yr 2000, and could be at repository site.	Discontinued development and heat transfer studies on metal transport/storage casks. Formerly used maximum fuel clad temp. of 250°C.	No experience to date.	Expect to store until at least 2020.
Former USSR	Demonstration of VVER fuel in Castor-V cask since 1984, and in a USSR transport cask since 1990, both inert gas. Demonstration of RBMK fuel (UO_2 with Zr clad) in air in hot cell.	Considering some dry storage, especially for RBMK fuel, in transportation/storage casks, in vaults, or in concrete casks.	Demonstrations of VVER fuel in Castor cask and in Soviet cask, and RBMK fuel tests in hot cell. Tests on degradation at various temperatures in air and inert gas.	VVER fuel in Castor-V cask since 1984, and in Soviet cask since 1990. Date unknown for RBMK fuel in hot cell.	No time period identified, but limiting temp. in air is 125°C and in inert gas is 350°C.

clad in magnesium-aluminum alloy) for only a few years (initially in CO_2 for about 150 days with a maximum cladding temperature of 365°C, then in air at a maximum cladding temperature of 150°C) before the fuels are sent to reprocessing. India has about 21 MTU of boiling-water reactor (BWR) fuel stored in four indigenous metal storage/transportation casks. All the other countries in Part A of Table 2.1 currently store only small quantities of specialty spent fuels.

Five of the six countries in Part B of Table 2.1 currently have no spent fuel in dry storage. The former USSR (most of the former republics in the former USSR are now in the Commonwealth of Independent States, CIS) has a few MTU of VVER (Soviet designation for its pressurized-water reactors [PWR]) fuel in demonstrations in two storage casks, and some RBMK (Soviet designation for its water-cooled graphite-moderated reactors) fuel in demonstration storage in a hot cell. These fuels are both UO_2 clad in zirconium. Italy has stored fuel from its Magnox reactor in sealed canisters in a storage pool, then in air for a few years, then in nitrogen. This fuel was shipped to the U.K. for reprocessing in 1990. Italy also has a small amount of specialty fuel stored in a hot cell.

As shown in Part A of Table 2.1, Canada, the Federal Republic of Germany (FRG before reunification with the former East Germany), Switzerland, and the U.K. plan to continue or start industrial-scale dry storage of spent nuclear fuels in the near future. The other three countries plan no such activities. Canada is planning to continue adding concrete cask storage of its CANDU fuels at a significant rate. The U.K. plans to continue its existing short-time (e.g., a few years) vault storage of Magnox fuels, and one utility is planning for future vault storage of fuels from its advanced gas-cooled reactors (AGR). The FRG has completed building away-from-reactor (AFR) storage facilities for dry storage of up to 3000 MTU of LWR fuels in transportable storage casks, to be used after litigation is satisfactorily concluded in the FRG. Switzerland is building an AFR storage facility that will use transportable storage casks. Vault storage and dry cask storage is licensed technology in those countries that are using or plan to use dry storage in the near future.

As shown in Part B of Table 2.1, Argentina, South Korea, and Spain each plan for near-future dry storage of hundreds of MTU of spent fuel in storage casks. Argentina (starting in 1993) and South Korea (starting in 1997) are preparing to use a Canadian-type of concrete cask for dry storage of their CANDU fuels in air. Spain is planning to use metal transportation/storage casks for extended interim storage of its LWR fuel in inert gas, pending licensing of the casks in the U.S. Significant amounts of spent fuel are planned to be in dry cask storage by 1995. The former USSR is considering the possible future use of dry storage casks or vaults, particularly for its RBMK fuel. There are indications that dry cask storage is now expected to be implemented for fuel from some of the now shutdown reactors in the former German Democratic Republic, and that some eastern European countries (e.g., Czechoslovokia, Hungary) are now planning for dry spent fuel storage.

Research work on dry storage in inert gases is considered to be basically completed in most of the countries, now that the technologies are licensed. The FRG conducted significant extensive research and development on dry storage of LWR fuels (primarily in inert gas atmosphere) in the early 1980s, and this work was satisfactorily concluded to the point of licensing the technology. Canada is continuing periodic long-term examination of its air-stored CANDU fuels and is continuing to develop its dry storage technology. Japan has carried out high-temperature oxidation tests of LWR fuels in air, the U.K. is starting research and development (R&D) on air storage of its AGR fuels, and the former USSR is continuing its three demonstrations and starting R&D testing on degradation of its VVER and RBMK fuels under various dry storage conditions. The other countries are generally not carrying out further R&D except perhaps for design/development work (such as heat transfer studies) for their specific cases. Canada is developing a concrete storage cask that can also be licensed for transportation, and is also developing dry storage vaults. The FRG is developing its "Pollux" cask for potential use for storage, transportation, and disposal of spent fuels that may not be reprocessed.

Industrial-scale dry storage in carbon-dioxide was first implemented in the U.K. for short-term storage of its Magnox fuels in 1972. Storage in air following about 150 days storage in carbon dioxide was started in the U.K. in 1979. Some examinations of U.K. dry-stored spent fuels have been carried out after about five years of storage, with favorable observations. (However, some Magnox fuel with cladding that failed due to corrosion from rain water leaking onto the fuel has been discovered recently in the U.K.'s storage vault. In addition, a small amount of Magnox fuel dry-stored in Italy for more than 15 years was found to have significant cladding damage, presumably due to high-temperature storage in an oxidizing gas.) In Canada, test storage in air using its concrete cask concept was started in 1980, and examinations have been made after 8.3 years of storage, with no degradation noted. The major R&D activities and extrapolations for the long-term storage of LWR fuels in the FRG in the early 1980s were done with fuel dry-stored up to two years. Japan has examined some of its dry-stored research reactor fuel after five years, with favorable observations. No other foreign observations of long-term dry storage are known.

In the FRG, the duration of the integrity of LWR fuels in dry storage in inert gas is expected to be in the order of 100 years, based on tests and predictions done in the FRG, the U.S., and elsewhere. This view is apparently shared by all the surveyed countries with interest in dry storage of LWR fuels. The duration of the integrity of defected LWR fuels stored in air is less well-defined: the Germans feel that oxidation testing is needed to define conditions and their effects on defected LWR fuel integrity in air storage of 40 to 100 years; the Japanese estimate that undefected LWR fuel can be stored for at least 30 years at 350 to 430°C, depending on fuel conditions and history, and defected LWR fuel can be stored for about 30 years in air at up to about 160°C. The Canadians feel that their cask air-storage concept at 100 to 150°C can easily store CANDU fuels for the 50 to 100 years that they may use dry storage, and that technical predictions indicate safe storage for 100 to 1000 years. Defected CANDU fuels have been stored without loss of integrity in air at 150°C (but with significant oxidation of the spent fuel at the grain boundaries) for eight years so far. The U.K. feels that it can store Magnox fuels in its vault concept for up to several decades.

2.2 CONCLUSIONS

Based on the findings in this survey, the overall conclusions that can be drawn for extended dry storage of spent fuels from nuclear power reactors are given below.

1. Foreign experience to determine long-term (i.e., decades) integrity of dry-stored LWR fuels has been limited mostly to technical tests and evaluations for about two years.

2. Based on past and current data developed in foreign countries and the U.S., inert gas storage of LWR fuels is considered to be well-developed by the countries surveyed, it is licensed for industrial application, and is being implemented in the market place by industry.

3. Because of the developed status of dry storage, relatively little R&D is in progress now on storage in inert gas in the countries surveyed. Most efforts are on development/design for specific applications. Active R&D efforts appear to be in progress in the former USSR for possible dry storage of their VVER and RBMK fuels; Canada is continuing long-term testing and fuel examinations for air storage of its CANDU fuels; and the U.S. and Japan are continuing some R&D on fuel oxidation in air storage.

4. The duration of LWR fuel integrity stored in inert gases at about 320 to 400°C is generally considered to be proven in the countries surveyed for the needed storage time periods of up to 50 to 100 years.

5. Storage times for LWR fuels in air must be shorter and at much lower temperatures than in inert gases, and there are relatively few results on long-term air-storage effects on defected spent fuels. Canadian tests have been carried out for 8.3 years on defected CANDU fuels in dry and moist air at 150°C with favorable results. The FRG experts feel more studies are needed to determine longevity in air storage for 40 to 100 years. Studies in the U.S. and Japan indicate allowable temperatures for air storage of LWR spent fuel are in the range of 135-160°C.

6. The U.K. was the country that first implemented industrial-scale dry storage (in 1972), but its storage is for the unique Magnox fuels and is for short time periods. Its experience has been favorable. Canada was the second country to implement industrial-scale dry storage (in 1980); its storage is at low temperatures in air for CANDU fuels. Canada's experience has also been favorable.

7. Canada and the U.K. are currently the only foreign countries with industrial-scale dry storage experience, and are among the most active in implementing their technologies. The FRG is also

planning to actively implement dry storage on an industrial scale as soon as the litigation on its away-from-reactor storage facilities is satisfactorily concluded. Several other countries are planning to implement industrial-scale dry storage in the near- to mid-term.

8. Casks (concrete and metal) and vaults are the most common dry storage concepts in use in the countries surveyed. All casks and most vault concepts in use utilize natural, convective air flow for cooling.

3.0 GENERAL BACKGROUND ON DRY STORAGE TECHNOLOGY

3.1 CONCEPTS OF DRY STORAGE

Dry storage of spent fuel from nuclear power stations has been carried out for small quantities of fuel and for tests since about 1960. Dry storage on an industrial basis has been used since 1972. This first industrial use was at the Wylfa nuclear power station in the U.K. for storage of Magnox fuels with a carbon dioxide cover gas, in a vault that was cooled passively by air. Two more of these vaults that use air as a cover gas have operated at the Wylfa nuclear power station since 1979 (IAEA 1988).

A number of concepts have been considered for dry storage of spent nuclear fuel, many of which have been implemented in selected countries. The DOE's initial research and development needs report for a Monitored Retrievable Storage (MRS) facility identified eight dry storage concepts that could be placed in three general categories (DOE 1983). These concepts are:

- Casks
  - Metal storage casks
  - Concrete casks (or silos)
  - Concrete casks in a trench or berm
- Drywells
  - Surface field drywells
  - Tunnel drywells
- Vaults
  - Surface, open-cycle vaults
  - Surface, closed-cycle vaults
  - Subsurface, open-cycle vaults (or tunnel racks).

In most dry storage concepts the spent fuel assemblies are oriented vertically, although some horizontal orientations have been used. The differences between the concepts within any category are generally not important from the standpoint of their impact on the longevity of the integrity of the spent fuel. For example, spent fuel integrity in casks is generally not impacted by whether or not the cask is metal or concrete or if the cask is exposed to the outside air atmosphere or is buried, unless other factors differ (e.g., storage temperature, storage gas composition [i.e., inert gas such as He or Ne or Ar or N_2, or oxidizing gases such as air or CO_2, or moisture content in the gas], failure of the integrity of the cask, etc.). Because of the significant effect of dry storage temperature on spent fuel integrity, heat transfer from the storage system is important. Dry storage concepts can use mechanically-forced heat transfer systems for improved heat transfer. However, existing installations typically use only natural passive external air cooling, which generally limits the spent fuels that can be stored to those that are several years out-of-reactor.

Industrial-scale storage has been implemented in several kinds of metal casks, in concrete casks (above-grade), and open-cycle surface vaults. All of these concepts transfer the decay heat within the spent fuel to the air in a passive, natural convection system. Subsurface dry storage has only been used for test purposes or for small amounts of materials (IAEA 1988). Prior reviews of all concepts for consideration in the U.S. MRS facility were discussed and published by Triplett and Smith (1984).

3.2 MECHANISMS FOR DEGRADATION OF SPENT FUEL CLADDING IN DRY STORAGE

Mechanisms for potential failure of the integrity of LWR spent fuel in dry storage have been identified as 1) cladding stress rupture due to the internal pressure of the fuel that creates high hoop stresses in the cladding wall, 2) stress corrosion cracking of cladding due to cladding interactions with fuel material contents such as iodine, cesium and cadmium; 3) external oxidation of the cladding material that could result in weakening of the mechanical characteristics of the cladding; 4) fatigue failure in the cladding from thermal cycling; 5) hydriding of the cladding material that could also weaken the mechanical characteristics of the cladding; and 6) splitting of the cladding due to oxidation and expansion of the UO_2 fuel. For storage in an inert gas, the most likely mechanism for cladding failure is creep rupture. For storage in air, cladding oxidation or splitting due to oxidation of exposed fuel materials are the most likely mechanisms for cladding failure. The other mechanisms are less likely to cause failures (IAEA 1985; EPRI 1989).

Maintaining the integrity of spent fuel depends on the storage conditions rather than on the storage concept. Controlled tests under various conditions have established design limits for extended storage. For LWR fuels, the maximum storage temperature limit is accepted internationally to be in the range of 320 to 400°C in an inert gas for LWR spent fuels clad with Zircaloy. The use of inert cover gases also helps to ensure that cladding defects do not increase during storage due to oxidation and resultant increase in volume of the fuel, thus causing splitting of the fuel rods. The acceptable temperature limit with oxidizing cover gases for Zircaloy-clad LWR fuels is probably in the range of 135 to 160°C (Cunningham et al. 1991; Einziger 1991; Einziger et al. 1992; Kawasaki and Nakamura 1991). Potential deleterious effects of dry storage on the integrity of spent fuel are generally limited by time-at-temperature. A simplified summary of the effects of various factors on the integrity of dry-stored spent fuel as of 1987 is given in Table 3.1. The table is derived primarily from information obtained by the IAEA from Canada, the Federal Republic of Germany, and the U.S. (IAEA 1988).

The most limiting degradation phenomena for dry storage of spent LWR fuels in inert gases has been shown by testing and modeling to be creep rupture resulting from internal fuel rod pressures and rod temperatures. The temperature criteria for extended storage without cladding failure due to creep rupture vary with the initial dry storage temperature, the long-term dry storage temperatures, and the out-of-reactor age of the fuel. The older the fuel when it is placed in dry storage, the lower the maximum allowable temperature during dry storage should be to ensure constant stress with time and to eliminate creep rupture within a given period of time. For a dry storage life of 40 years for 5-year-old fuel in inert gases, the allowable initial dry storage temperature, as determined by U.S. testing, is about 380°C; for 15-year-old fuel, the allowable initial dry storage temperature is about 350°C. For longer dry storage periods, the initial dry storage temperature would be somewhat lower (Gilbert et al. 1990).

A study carried out by the Electric Power Research Institute (EPRI) (1989) assessed the probability of LWR fuel rod cladding failure versus dry

TABLE 3.1. Overview of Effects of Conditions on Integrity of Spent Fuel in Dry Storage (IAEA 1988)

Factors Affecting Spent Fuel During Dry Storage	Effects Using Inert Gas	Effects Using Oxidizing Gas
Pre-storage in a water pool	None if temperatures remain below design limit	Same as with inert gas, but design temperature limit is lower
Condition of fuel assemblies		
- Fuel type	None	None
- Clad type: Zr-2, Zr-4, Zr-2.5Nb	None	None
- Pressurization	Minor effect	None
- Operational defects	None	Guiding criterion
- Crud	Only affects retrievability	Only affects retrievability
Dry storage conditions		
- Temperature/type	Guiding criterion	Limited by defect behavior
- Atmospheric impurities	Minor (if water)	Possible synergistic effect with water
- Storage-induced defects	Seldom	Less limiting than operational defects
- Packaging	None	None
Retrievability		
- Time <10 yrs	Easily retrievable	Easily retrievable if operational defects are excluded or if low frequency of UO_2 oxidation
- 10 to 40 yrs	Easily retrievable	
- Time >40 yrs	Easily retrievable	

storage temperature by several mechanisms over a dry storage period of 100 years. The study concludes that:

- "If a 'high' probability of failure (such as 0.01 rod failures per rod in 100 years) is acceptable and storage is in air, then external oxidation dominates as a failure mechanism and allowable temperatures are around 325°C."

- "If a 'high' probability of failure per rod is acceptable and storage is in inert atmospheres, then creep rupture dominates as a failure mechanism and allowable temperatures are a bit higher, around 350°C."
- "If, however, only a 'low' probability of failure (such as 0.0001 rod failures per rod in 100 years) is acceptable, then stress-corrosion cracking dominates, and allowable temperatures are calculated to be in the neighborhood of 150°C."

Thus, when following the criteria discussed above, failure of the LWR cladding in some small amount of the spent fuel rods (i.e., one percent in 100 years) can be expected during dry storage (EPRI 1989). In another study, the probability for failure of Zircaloy cladding due to creep rupture at 380°C is estimated to be less than 0.005 rod failures per rod over a 40-year storage period (Cunningham et al. 1987).

Dry storage of spent LWR fuel in air has not been approved in the U.S. because the database on the effects of air on spent fuel integrity is not yet adequate to establish specific allowable temperature limits. Currently, the temperature limit from testing in the U.S. is expected to be in the range of 135 to 150°C (with the air dew point no higher than about 40°C) for 40 years storage, and somewhat lower for longer storage periods. The concern with storage in air is oxidation (and possibly hydration) of the UO_2 fuel through an unknown cladding defect to a less dense oxide (e.g., U_4O_9 and an amorphous phase) that will increase the local fuel volume, resulting in growth of the cladding defect and potential release of the fuel material. Testing is under way in the U.S. to allow for future modeling and prediction of oxidation of the irradiated UO_2 with variations in temperature and moisture content. (Gilbert et al. 1990; Cunningham et al. 1991; Einziger 1991; Einziger et al. 1992).

3.3 MULTI-NATIONAL ACTIVITIES IN DRY SPENT FUEL STORAGE

The International Atomic Energy Agency (IAEA) has a continuing program on spent fuel management. To assist the IAEA in steering its spent fuel study activities, a Regular Advisory Group on Spent Fuel Management (RAG-SFM) meets approximately every two years to review the IAEA's spent fuel program activities and to advise them on future activities. The RAG-SFM also uses these biannual meetings to hear presentations from member countries on the status of their spent fuel management activities (IAEA 1990).

The IAEA carried out a coordinated research program (CRP) on Behavior of Spent Fuel Assemblies During Extended Storage (BEFAST-Phase I) from 1981 to 1986. The CRP consisted of experts from 10 countries (Austria, Canada, Czechoslovakia, Finland, the former German Democratic Republic, Hungary, Japan, Sweden, the U.S., and the former USSR) that periodically met and presented results of research on spent fuel storage in their countries. The aim of the program was to provide a significant database on the cladding integrity of spent fuel after extended storage periods. Technical objectives were to 1) survey existing experience in spent fuel storage; 2) investigate spent fuel by destructive examinations before and after extended storage periods; 3) investigate potential cladding degradation mechanisms; 4) evaluate suitable nondestructive testing methods for surveillance; and 5) to investigate the behavior of spent fuel pool equipment. The results of BEFAST-I were reported in IAEA-TECDOC-414 (IAEA 1987), and are included in this study.

The second phase of the IAEA's BEFAST CRP (BEFAST-II, or Behavior of Spent Fuel and Storage Facility Components During Extended Storage) was carried out from 1986 to the final meeting in March 1991. BEFAST-II had similar objectives to BEFAST-I and included member participants from 12 countries (the 10 in BEFAST-I plus the Federal Republic of Germany and Japan). The results of the BEFAST-II will be documented in an IAEA report, but some of the information is cited in this report.

The Commission of the European Communities (CEC) carried out a cost-sharing research program with member states on "Long-Term Storage of Spent Nuclear Fuel." The final report on the program was published in 1988, and the results of this program were also reviewed for this study (Fettel et al. 1988).

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4.0 DRY STORAGE EXPERIENCE IN CANADA

4.1 NUCLEAR POWER, REACTORS, AND SPENT FUEL

Canada has had nuclear power since starting up its demonstration power reactor in 1962 and its first industrial-scale reactor in 1968. Since then, Canada has implemented a steady growth of nuclear power (Frost 1988). At the end of 1991, 19 nuclear power reactors were in operation in Canada, with a gross electrical generation capacity of 13.9 GW. The capacities of individual reactors vary from 540 to 915 gross MWe. Three older reactors have been shut down, three new reactors are under construction, and a fourth is planned (NEI 6/90; NW 2/6/92).

Canada has developed indigenously the Canada Deuterium Uranium (CANDU) reactor system, which is used for all nuclear power generation of electricity in Canada. Canada also exports the CANDU reactors to other countries. The reactors are moderated and cooled with pressurized heavy water, with secondary coolant of light water in the steam generators. The reactor vessel, or calandria, contains numerous horizontally-oriented tubes, each of which contains 12 to 13 fuel bundles in series. A reactor contains 4560 to 6240 fuel bundles, with a total fuel loading of 95 to 108 MT of UO_2 (Nuclear Canada Yearbook 1983). CANDU reactors are refueled on-line, and each reactor (greater than 500 MWe capacity) discharges an average of 10 or more fuel bundles every operating day (Frost 1988).

The spent fuel from CANDU reactors is natural uranium dioxide (UO_2) pellets in Zircaloy tubes. A typical fuel assembly consists of 19 to 37 tubes (arranged in a circle; the 37-tube assembly has a center tube and three concentric circles containing 5, 12, and 18 tubes, respectively) containing about 20 kg of uranium. Each assembly is about 0.5 m long by about 10.2 cm in diameter. Each tube is made from Zircaloy-4. The tubes in the 37-tube assembly average 13.07 mm in diameter and have a wall thickness of about 0.419 mm. All CANDU reactor fuel fabricated since 1974 has a thin layer (about 20 μm thick) of graphite on the inside of the tubes to reduce stress corrosion cracking and embrittlement due to fission products. Burnup in the reactors, which averages about 7500 MWd/MTU, results in a spent fuel composition containing about 0.22% ^{235}U, 0.38% Pu, and 0.76% fission products. The reference heat content at 10 years out-of-reactor is about 218 W/MTU. The freshly discharged spent fuel is stored horizontally in baskets, modules (a compact basket), or trays in the reactor pools. The module will also serve as a transportation basket and thus will eliminate an additional handling step (Frost 1988).

4.2 SPENT FUEL MANAGEMENT POLICY

In Canada, the interim storage of spent nuclear fuel is the responsibility of the generator of the spent nuclear fuel. Because a repository for spent fuel is not expected to be available for at least another 30-40 years, extended retrievable storage is required. Spent fuel is stored in the reactor primary pool, then moved to an auxiliary storage facility. The spent fuel generators plan to expand their storage capacity by at-reactor storage or storage facilities that service a multiple-reactor station. The expanded interim storage was originally expected to be either wet or dry storage. [However, the current trend for future expanded storage appears to be dry storage in concrete casks, or possibly in vaults.] Canada currently does not plan to reprocess the spent fuel, but the final decision on reprocessing will be made by the time the disposal repository for spent fuel or high-level waste (HLW) is available (Schneider et al. 1991).

Two strategies for storage of spent fuel are being developed in Canada: 1) interim storage for the "near term" of about 50 years and 2) extended storage for a longer term of about 100 years if decisions are delayed on the ultimate disposition of the spent fuel. The Canadians believe that they have developed storage technologies to safely and effectively carry out these strategies (IAEA 1990).

4.3 STATUS OF SPENT FUEL STORAGE

All CANDU reactors in Canada discharge spent fuel directly into the water pools at the reactors. After interim storage in the primary pool for several months to about seven years, the spent fuel is transferred to an auxiliary wet or dry storage facility (Frost and Wasywich 1991a).

Canada has studied dry spent fuel storage at the Whiteshell Nuclear Research Establishment (WNRE) since 1975. Spent fuel storage in a dry air atmosphere has been licensed and implemented as supplementary storage to in-pool storage in Canada for about eight years. To date, supplementary storage in new pools has been implemented at only one reactor station for extended storage. The favorable Canadian experience with wet storage has led them to predict maintenance of CANDU fuel integrity for at least 50 years while stored in pools. By the end of 1991, Canada had more than 600 MTU of CANDU spent fuel in licensed dry storage at five facilities. All dry storage facilities use steel-lined concrete storage casks (also called canisters) of similar design but with evolutionary differences. Dry air storage in concrete canisters has been found to be satisfactory for the CANDU fuel where storage temperatures are low enough to prevent rapid degradation of the cladding or UO_2. Maximum fuel cladding temperatures and maximum allowable concrete temperatures limit the storage to fuel that is at least five years out-of-reactor (IAEA 1987; IAEA 1988a; Frost and Wasywich 1991c).

Ontario Hydro and Atomic Energy of Canada Limited (AECL) have a long-term program, initiated in 1977, to examine spent fuel stored in their water storage pools for possible deterioration. The oldest fuel bundles selected for the program have been in wet storage since 1962. Results of tests in 1978-1979 on pool-stored CANDU fuel bundles indicate that no degradation of fuel cladding or the UO_2 fuel (for defected fuel cladding) should result from water storage for at least 50 years. Examinations conducted in 1988-1989, after an additional 10 years of storage, also supported that result. All examinations indicated no significant change in the fuel element integrity and the condition of the cladding after 27 years of wet storage, and the UO_2 matrix in defected fuel after up to 21 years of wet storage (Frost and Wasywich 1991a; 1991b).

Since implementing the use of a thin lining of graphite inside the CANDU fuel rods in 1974, the rate of leaking fuel assemblies during reactor operation been reduced from "high" rate of 1% to the "low" rate of less than 0.01% (Frost 1988; IAEA 1990). [Thus, the use of the graphite tube lining has reduced the failure of the cladding from some mechanism, which the Canadians seem to think was related to the presence of the fission product iodine.]

4.4 EXPERIENCE IN DRY SPENT FUEL STORAGE

Dry storage concepts have been studied in Canada for possible long-term storage (e.g., 50 to 100 years or so) of spent fuel since 1975. Concepts considered include concrete canisters (or casks), convection vaults, concrete integrated containers (CIC), and cast iron casks. The concept implemented to date is that of licensed concrete canisters. In addition, a demonstration program is well along on concrete integrated containers, which are concrete casks that can also be used for transportation. The concrete canisters provide two engineered barriers to release of spent fuel materials to the environment (described below): the sealed storage basket that holds the fuel assemblies and the inner steel liner of the concrete canister (or cask), whose lid is sealed shut by welding to the liner (Patterson and Hoye 1986; Truss and Patterson 1987; Frost 1988).

Because of excellent performance in early tests, the AECL has decided to use concrete canisters to store all irradiated fuel from the Whiteshell WR-1, Gentilly-1, Douglas Point, and Nuclear Power Demonstration reactor at Chalk River. In addition, the New Brunswick Electric Power Commission decided to store fuel from the Point Lepreau nuclear power station in concrete canisters. By the end of 1991, 104 canisters were to store fuel bundles containing a total of about 606 MT from these five reactor sources in licensed facilities in Canada as follows (Frost and Wasywich 1991c):

- About 24 MTU of the AECL's WR-1 (research reactor) fuel (with slightly enriched uranium) is stored in 14 cylindrical concrete canisters (2.5 m outside diameter, 5.5 m high) with an inner steel liner. [Some of the early casks had lead liners for shielding between the steel liner and the concrete.] About 10 MTU have been in dry storage since 1977. The vertically-oriented fuel assemblies in each canister (24 to 37) are in six sealed stainless steel baskets backfilled with helium, with dry air outside the baskets. Each cask has a metal cap over the top to act as a weather shield. The casks are licensed for a maximum thermal rating of 4.4 kW, and up to 6 MT of uranium (Patterson and Hoye 1986; Smith and Zarecki 1987; IAEA 1988b; Frost and Wasywich 1991c).

- Hydro-Quebec's Gentilly-1 fuel (3213 assemblies, 67 MTU) has been stored in 11 WNRE-type concrete casks inside a building since 1985. Each steel-lined cask is 2.6 m in diameter and 6 m high, with a steel-lined cavity 0.78 m in diameter and 4.6 m high. Each cask has eight sealed stainless steel canisters filled with 38 vertically-oriented spent fuel assemblies and is backfilled with an air atmosphere. Each basket was dried internally using hot air, after loading under water and before sealing at atmospheric pressure. The casks can hold 459 assemblies (about 9 MTU) (Patterson and Hoye 1986; Smith and Zarecki 1987; Frost and Wasywich 1991c).

- About 65 MTU of spent fuel from the AECL's Nuclear Power Demonstration Reactor at Chalk River has been stored since 1987 in 11 steel-lined concrete canisters that are similar to those for Douglas Point fuel. The 9 MT of this fuel that is enriched up to 1.4% U-235 is stored in 48-assembly baskets instead of the 54-assembly baskets for natural uranium. Nine baskets are in each concrete storage canister (Wasywich and Frost 1989b; Frost and Wasywich 1991c).

- The AECL's Douglas Point fuel (23,000 assemblies, or 300 MTU from the shutdown CANDU reactor) has been stored in 46 above-ground, steel-lined concrete casks in sealed stainless steel canisters in an air atmosphere. About 5 MTU of this has been in dry storage since 1976. These canisters are similar to those used at Chalk River. The casks are 6.16 m high, with a cavity of 0.84 m in diameter. Each cask can hold 486 fuel bundles (about 6.5 MTU) in nine sealed stainless steel fuel baskets holding 54 vertically-oriented fuel assemblies each (IAEA 1988b; NN 1/88; Frost and Wasywich 1991c).

- The New Brunswick's Electric Power Commission's Point Lepreau station had 15 canisters containing about 150 MTU in storage by the end of 1991. The canisters are similar to those at Douglas Point, but each of the nine baskets holds 60 fuel bundles. The canisters have an inside diameter of 1.12 m, an outside diameter of 3.0 m, and are 6.4 m high. The facility will have capability for storing up to 300 canisters. Loading of these large canisters (which can each contain 10.3 MTU fuel that is at least six years out-of-reactor) started in late 1990 (Alikhan 1990; Frost and Wasywich 1991c).

The reference dry storage facility design in a demonstration program by Ontario Hydro in Canada is a rectangular concrete container, Dry Storage Canister (DSC) that uses passive air cooling of irradiated fuel that is at least 6 to 10 years out-of-reactor (fuel temperatures no higher than 100°C). The DSC is a later, upgraded version of the cylindrical concrete integrated container (CIC) under development from 1981 to 1988. The DSC, which is somewhat similar to AECL's concrete canisters and has been under development since 1981, is intended to be a container that can be used for storage, transportation, and possibly disposal of Ontario Hydro's spent fuel. The DSC is a high-density concrete cask that is lined internally and externally with mild steel. Its outside dimensions are 2.12 m by 2.42 m by 3.55 m high, and it weighs 58 MT empty. It will hold four storage modules with a total of 384 fuel bundles (7.7 MTU, 8.88 MT mass) oriented horizontally in the vertically-oriented cask. It is sealed with a welded-on, steel-encased concrete lid. Two prototype PSCs have been loaded with fuel that was 6 and 10 years old at the time of loading at the Pickering Site (Frost 1988; Stevens-Guille 1991).

In the future, Ontario Hydro will start using its DSC to store fuel from its Pickering A and B reactors. Loading of Pickering fuel is expected to begin in 1993-94 at a rate of 500 MTU/year. The building for storing the casks will hold all spent fuel discharged until 2006 (about 6200 MTU in 700 casks). After that, another building may be constructed as a second phase for more concrete canisters to store an additional 7,000 MTU. Each canister will hold 384 CANDU fuel bundles (Stevens-Guille et al. 1992; NW 1/16/92). The New Brunswick Electric Power Commission is planning to load about 10 concrete storage canisters with spent fuel each year at Point Lepreau for about 30 more years for an eventual total of 2790 MTU in 273 canisters. Hydro Quebec is evaluating the use of dry storage for more of its Gentilly-2 CANDU fuel. Tentative plans include loading spent fuel into concrete canisters at the rate of 100 MTU/year starting in 1994 (Frost and Wasywich 1991c).

AECL plans to continue development of dry storage technology. The stand-alone concrete canisters will be replaced by ventilated concrete vaults, called CANSTOR modules. AECL's Whiteshell Laboratories is in the process of obtaining nuclear criticality approval to load about 1 MTU of enriched fast-neutron fuel into concrete canisters (Frost and Wasywich 1991c; Pattantyus, 1991).

In 1989, the AECL and Transnuclear developed an air-cooled vault concept (MACSTOR) for storing canisterized LWR fuel. In 1990, the concept was adapted for storing canisterized CANDU fuel (CANSTOR). Thermal testing of the natural-draft-cooled vault concept has been completed and a system has been designed. The two systems are designed for use as at-reactor storage systems (Pattantyus 1992).

The AECL has signed an agreement with the Korean Electric Power Corporation (KEPCO) to provide AECL's dry storage technology for spent fuel in the CANDU Wolsung-1 reactor (Frost and Wasywich 1991c; Park et al. 1991).

4.5 R&D IN DRY SPENT FUEL STORAGE INTEGRITY

AECL and Ontario Hydro have been conducting three experiments since 1978 to investigate the behavior of CANDU fuel bundles in dry cask storage. The three experiments are 1) the Easily Retrievable Basket (ERB) experiment in dry air at seasonably varying temperatures; 2) the Controlled Environment Experiment (CEX), Phase 1, with eight fuel assemblies in dry air at 150°C; and 3) the CEX-2, Phase 2 experiment in air saturated with moisture at 150°C. The three experiments are planned to be continued beyond the year 2000 (Frost 1988; Wasywich and Frost 1988).

The ERB experiment involves storage of undefected CANDU fuel and WR-1 fuel bundles in an air atmosphere. Storage temperatures have ranged from 55°C to -19°C. The fuel was examined in 1982 and in 1988, with no detectable change in its condition. The next examination is scheduled for 1993 (Crosthwaite 1990).

Results of the Controlled Environment Experiments in 1988 showed no loss of cladding integrity in any experiment after 58 months of the test, and some fuel grain boundary oxidation but no loss of integrity of UO_2 in defected fuel rods, at both dry and moisture-saturated air at 150°C (Frost 1988).

A second examination was performed on an intentionally-defected CANDU spent fuel assembly stored in dry air at 150°C in a concrete canister at Whiteshell as part of Phase 1 of the Controlled Environment Experiment. The assembly had been irradiated in the Bruce-A nuclear reactor for an average of 8000 MWd/MTU from July 1976 to December 1977; it was stored under water for 1.6 years and had a decay heat of 23 watts when it was placed in dry storage. The assembly was placed in dry storage in 1980 and was first examined in

1984 after 41 months of dry storage. In 1984, a small amount of oxidation of the fuel was found, but no increase in fuel rod diameter was observed, and no movement of radioisotopes was detected. In April 1989, the same intentionally-defected fuel assembly was retrieved for examination after 99.5 months in storage. A significant increase in the amount of fuel experiencing oxidation was observed (overall ratio of U^{+6}/U^{+4} increased from 0.24 after 41 months to 0.8) since the first examination. Storage of the bundle is continuing and future examinations are planned (Wasywich and Frost 1991).

Retrieval and examination of an intentionally-defected CEX phase 2 test fuel assembly was carried out in June 1984, after 30 months of storage in saturated air at 150°C. The test, carried out in a stainless steel pressure vessel, was saturated with distilled water (at pH of 5.5) at the test temperature and 5.3 atmospheres of pressure. A significant amount of grain boundary oxidation occurred at the defect. However, no increase in fuel rod diameter was observed, indicating the amount of oxidation did not affect the structural integrity of the fuel rod. In April 1988, the same fuel assembly was retrieved after 69 months of storage in moisture-saturated air and examined the second time. (Because all the oxygen in the pressure vessel had been used up as of April 1988, the subsequent experiment, now in progress, was redesigned to provide an unlimited supply of saturated air.) The amount of fuel in the defected fuel rods that experienced grain boundary oxidation varied from about 50 to 65% after 69 months, and generally was higher for fuels with higher irradiation levels. The releases of radiocesium nuclides to the liquid water in the test pressure vessel were 0.03 to 0.06% of the total inventory in the defected fuel rods. The dimensions and integrity of the fuel rods were unchanged (Wasywich et al. 1986; Wasywich and Frost 1989a). Additional observations have been made on intentionally-defected fuel bundles removed in December 1990 after storage periods of 86 to 93 months. Oxidation of fuel along the grain boundaries had grown, but no evidence of bulk oxidation of grains was observed. The releases of radiocesium and tritium to the storage water in this case were 0.03% and 1%, respectively (Wasywich and Frost 1992).

In the fall of 1990, Ontario Hydro loaded a second DSC with 384 assemblies (7.7 MTU) of six-year-old Pickering-A spent fuel as part of the ERB experiments. This container and the prior test unit (with 10-year-old fuel) were loaded in the spent fuel interim storage pool, transferred by the pool crane to a trailer, then moved to an outside storage area. Ontario Hydro has carried out preliminary drop and fire tests on a 1/4-scale rectangular dry storage container for transportation licensing purposes, with favorable results. Drop tests were to be repeated on a 1/2-scale container in 1991 (Frost and Wasywich 1991c; Stevens-Guille 1991).

4.6 PROJECTIONS OF LONGEVITY OF SPENT FUEL IN DRY STORAGE IN CANADA

Using Arrhenius plots for CANDU fuel UO_2 oxidation, the cladding of CANDU fuel bundles is predicted to lose no integrity during storage in air at 100°C or less for 100 to 1000 years (Frost 1988; Stevens-Guille 1991).

To ensure that the integrity of spent fuel can be maintained for 50 years or more in dry storage, with allowance for possible defected spent fuel, the Canadians recommended that their containers and the spent fuel bundles be dried as thoroughly as possible prior to dry storage (Wasywich and Frost 1988). The design life of the Dry Storage Canister is 50 years (Alikhan 1990).

Although no formal prediction is given in the Canadian literature, the existing data for storage of defected fuel in dry air at 150°C after 8.3 years (with essentially no effect on the fuel integrity), it has been stated that "undefected and defected used CANDU fuel elements can be stored safely in a moisture-saturated, limited-air environment at 150°C for at least 93 months without experiencing any significant degradation." (Wasywich and Frost 1992.) However, one would expect to see important degradation of such fuel assemblies (due to UO_2 oxidation) after storage of several tens of years. Similarly, important degradation will occur sooner in moist air at 150°C than in dry air; the examination after 69 and 93 months of storage in moist air at 150°C resulted in grain boundary oxidation of more than half the fuel in defected fuel rods.

The research done at Whiteshell indicates the peak air temperature reached during concrete canister storage of Douglas Point fuel, about 110°C, was acceptable for storing the fuel baskets in dry air. After 10 years of storage, the maximum temperature dropped to about 66°C. The corresponding peak temperature for freshly-loaded air storage of the Gentilly-1 spent fuel was less than 60°C (Wasywich and Frost 1988).

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5.0 DRY STORAGE EXPERIENCE IN THE FEDERAL REPUBLIC OF GERMANY (FORMER WEST GERMANY)[a]

5.1 NUCLEAR POWER, REACTORS, AND SPENT FUEL

The Federal Republic of Germany (FRG) started nuclear power production in 1962 at its Kahl VAK reactor (16 MWe), followed by startup of its first industrial-scale reactor in 1966 at Gundremmingen (250 MWe). It presently has 22 nuclear power reactors in operation with a gross electrical generation capacity of 23,636 MWe. The FRG operating nuclear reactors include 14 pressurized water reactors (PWRs), seven boiling water reactors (BWRs), and one small fast breeder reactor (FBR) (20 MWe, at Karlsruhe). The capacities of the individual light water reactors (LWRs) vary from 357 to 1390 gross MWe. Seven older reactors and one fairly new reactor [four BWRs, one gas-cooled heavy water moderated reactor (GCHWR), two high-temperature gas-cooled reactors (HTGRs), and one pressurized heavy water reactor (PHWR)] have been shut down. The THTR-300 thorium high-temperature reactor (HTGR-type) was only operated from 1987 until 1990. One FBR (SNR-300, at Kalkar) is presently under construction, and 10 reactor units have been cancelled or indefinitely deferred. The FRG supports using nuclear energy as an important part of a diversified energy supply. It supports continued reactor development in the form of improved LWRs, FBRs, and HTGRs (NEI 1991; Schneider et al. 1991).

The spent fuel in the FRG consists primarily of LWR fuels, both PWR and BWR spent fuels, but significant amounts of HTGR, GCHWR, PHWR, and FBR spent fuels also exist. The LWR fuels are clad with Zircaloy. The PWRs use typical fuel assemblies, ranging in pin array size from 14 x 14 to 18 x 18 pins in a square array in each assembly. Most of the BWRs use assemblies with an 8 x 8 pin array, but the newer BWRs have pin arrays of 9 x 9. The PWR spent fuel planned for dry storage (specified for the Gorleben AFR facility) has a

(a) This section covers only the Federal Republic of Germany as it existed before unification with the Former German Democratic Republic in October 1990. Note that the current unified Germany is also the Federal Republic of Germany.

typical burnup of 35,000 MWd/MTU (enrichment 3.2% ^{235}U) and the BWR spent fuel has a typical burnup of 33,000 MWd/MTU (enrichment 2.8% ^{235}U) (Schneider et al. 1991; Muller 1986).

The HTGR fuel in the FRG reactors, referred to as "pebble-bed" reactors, consists of spherical fuel elements, each about 6 cm in diameter and clad with graphite. The fuel elements contain pyrocarbon-coated high-enriched uranium and thorium microspheres (0.76 mm diameter). The THTR-300 reactor contained 675,000 spherical UO_2/ThO_2 fuel elements (0.3 MTU, 6.6 MTTh) that have an average burnup of 114,000 MWd/MTHM when discharged from the reactor. The AVR reactor contained 106,000 mixed oxide-carbide fuel elements (0.62 MTHM) with an average burnup of 150,000 MWd/MTHM (IAEA 1988).

5.2 SPENT FUEL MANAGEMENT POLICY

The spent fuel management strategy in the FRG is based on interim storage for about one to ten years and subsequent reprocessing of spent fuel. The high-level wastes resulting from reprocessing are to be vitrified and disposed of in a deep geological repository (at Gorleben, if site suitability is confirmed). The direct disposal of spent fuel without reprocessing may be used for spent fuel that is not suitable for reprocessing (i.e., technically not feasible or economically not viable) (Peehs and Rudolf 1991).

In the FRG, the federal government is responsible for construction and operation of final disposal repositories for radioactive wastes. Industry is responsible for all other facilities such as interim storage facilities for spent fuel assemblies at-reactor (AR) or away-from-reactor (AFR), reprocessing facilities, and conditioning facilities for spent fuel assemblies if it is not technically or economically feasible to reuse (Johnson 1989).

Utilities are required to demonstrate provisions for spent fuel management to obtain a license to operate a reactor. Before the first partial operating license is granted, the utility must demonstrate that the safe location of spent fuel is ensured for six years after plant commissioning. This is provided by onsite and offsite interim storage, followed by reprocessing of spent fuel and conditioning of the radioactive wastes for disposal. Spent fuel is to be stored at reactors for a minimum of one year and typically five to ten years; offsite storage is to be provided until reprocessing or disposal is accomplished (Schneider et al. 1989; Schneider et al. 1990).

Interim storage of spent fuel is performed at the nuclear power plants and in AFR interim storage facilities, prior to its transport to foreign reprocessing facilities (in France and the United Kingdom). Onsite storage of spent fuel at reactors consists of wet storage in spent fuel pools in compact racks. Consolidation of spent fuels is also being developed for possible future use to increase the present storage capacities (Johnson 1989). Storage of spent fuel at AFR facilities will be conducted using dry storage casks (dual-purpose transport and storage casks) (Peehs and Rudolf 1991). The minimum cooling time (in AR pools) required prior to dry storage is about one year. AFR facilities have been constructed at Ahaus and Gorleben (construction at Gorleben was completed in 1984), but their use has been delayed by litigation. Use of the Gorleben AFR facility was approved in 1991 after seven years of litigation (NF 3/4/91). The vitrified high-level wastes returned from reprocessing are also planned to be stored in dry casks at AFR sites (Schneider et al. 1990).

The spent fuel for which reprocessing is considered not to be practical comprises about 20% of all spent fuel and consists of spent fuel from HTGRs, FBRs, mixed-oxide (MOX) fuels in LWRs, and other special fuel types (Schneider et al. 1989). These types of spent fuel are planned to be disposed of directly without reprocessing. A pilot conditioning plant (PKA), with a throughput of 35 MT/yr, is being constructed (with startup scheduled for 1994) at Gorleben to develop conditioning techniques for direct disposal of spent fuel (Johnson 1989). Spent fuel from HTGRs will be sealed in stainless steel canisters and stored until disposal, with about two years in pool storage followed by dry storage (Schneider et al. 1991).

5.3 STATUS OF SPENT FUEL STORAGE

Approximately 5600 MT of spent fuel storage capacity is available at the nuclear power plants, and 3000 MT of spent fuel storage capacity is available at the two AFR facilities (1500 MT each at Gorleben and Ahaus). Storage of LWR spent fuel in water using standard or high-density racks is considered to be a mature and viable technology; experience has shown that zirconium alloy (Zircaloy) clad fuel has not deteriorated to date (Peehs and Rudolf 1991).

The FRG has decided to use dry storage in dual-purpose (i.e., transport and storage) casks as its concept to extend the interim storage capacity for spent fuels. The dry storage facility at Ahaus consists of a large, naturally-ventilated building with space to store, in 320 dual-purpose casks, 1500 MTU of LWR fuel as well as 608,000 fuel assemblies from the THTR at Hamm-Uentrop. The Gorleben facility is similar, with storage capability for 1500 MTU of LWR fuel and vitrifed high-level waste from foreign reprocessing (Heimlich 1991). Dry storage has been demonstrated at reactor sites and by loading LWR fuel into dual-purpose casks, followed by shipment to and storage at R&D sites. The plan to implement dry storage in casks at AFRs has been on hold due to political litigation, and although it was approved at one of the AFRs in 1991, it has not yet been used (NF 3/4/91). Results (described in Section 5.5) of dry storage demonstrations of casks and the spent fuel behavior are considered by the FRG to confirm the maturity of this technology so that further cask development will concentrate on optimization, to be carried out by the cask suppliers (Einfeld and Popp 1986; Muller 1986).

A dry storage facility for HTGR (AVR reactor) spent fuel with a capacity of 0.62 MT is in use at KFA (Julich Research Center), and there have been no problems. At-reactor dry storage for spent THTR fuel has been used since 1988. Transport/storage casks (similar to those for LWR fuel) have been selected for further storage of both AVR and THTR spent fuel at the Ahaus AFR (IAEA 1988; Kirch 1990; Heimlich 1991).

5.4 EXPERIENCE IN DRY SPENT FUEL STORAGE

5.4.1 Spent Fuel Dry Storage Cask Designs

Based on extensive experience with transport casks, several types of dual-purpose (i.e., transport and storage) casks have been developed, licensed, and manufactured in the FRG. GNS (Gesellschaft fur Nuklear Service, Company for Nuclear Service) and TN (Transnuklear) are the suppliers of these casks. DWK (Deutsche Gesellschaft fur Wiederaufarbeitung von Kernbrennstoffen, German Fuel Reprocessing Company), which previously had responsibility for the AFR storage facilities, has tested several casks and conducted procedures to support licensing of the casks for interim storage. GNS has taken over responsibility for the Gorleben and Ahaus AFR facilities (Schneider et al. 1990; Einfeld and Popp 1986; Muller 1986; Leigh and Mitchell 1990).

GNS Castor casks consist of a ductile, nodular cast-iron body (wall thickness of about 40 cm) cast with cooling fins at the outer surface (longitudinal on Castor I series, circumferential on Castor V). The internal cask surface is covered by a corrosion-resistant material (nickel plating or a stainless steel liner). The Castor series of casks has a maximum capacity of up to 10 MTU in spent fuel (up to 21 PWR assemblies), a weight of 60-120 MT, and a design thermal capacity of 30 to 50 kW. They are equipped with two bolted-on lids, each with a gasket sealing system (using nickel-base alloy Helicoflex gaskets) to ensure long-term tightness, with leak-monitoring equipment. After loading, drying, and filling with helium, the casks are transported to a dry storage facility (Spilker and Fleisch 1986; Muller 1986; Droste 1985; Droste 1986; Schneider et al. 1990).

Castor Ia and Ib casks are designed to contain 4 PWR fuel elements, Castor Ic casks contain 16 BWR fuel elements, Castor IIa casks contain 9 PWR fuel elements, and Castor KRB-MOX casks contain 16 mixed-oxide fuel elements. Castor Ic and V/21 casks were the first casks approved by the U.S. Nuclear Regulatory Commission for dry spent fuel storage in the U.S (DOE 1989).

The Castor casks are licensed for an intermediate storage period of 40 years in the AFR facilities. Fuel is to be stored in helium gas at about one atmosphere pressure (IAEA 1988). Based on the creep rate of Zircaloy cladding, a maximum allowable cladding temperature of 410°C was derived for the AFR licensing process (Einfeld and Popp 1986; Muller 1986).

The TN casks are fabricated from either cast iron or forged carbon steel. The TN-1300 cask (cast ductile iron) has a maximum capacity of 12 PWR or 33 BWR assemblies, a weight of 116 MT, and a design thermal capacity of 50 kW. It uses double metallic O-rings in the cask sealing system, with leak-monitoring equipment, and has longitudinal exterior fins for cooling purposes. The TN-24 storage cask has a forged carbon steel body and is designed for five-year cooled spent fuel. It is designed for a maximum fuel clad temperature of 375°C. The TN-24 cask has a smooth outer surface, with interior fins for heat conduction to the outer surface. It has a design capacity for 24 PWR or 52 BWR spent fuel elements. The TN-1300-1/9 cask is designed to contain 9 PWR fuel assemblies, the TN-1300-1/12 contains 12 PWR fuel assemblies, and the TN 900 cask contains 21 BWR assemblies (Pennington 1986; DOE 1989; Schneider et al. 1990).

The Pollux cask system, being developed by GNS, is a triple-purpose (i.e., transport, storage, and disposal) cask system designed for spent fuel that will be disposed of directly. The Pollux cask system currently consists of a sealed inner cask with a Hastelloy-C4 coating (8 mm thick) for corrosion protection during disposal and an overpack (not leak tight) for shielding, handling, and transport. Consolidated fuel rods from eight PWR assemblies are enclosed in thin-walled canisters and positioned in the perimeter of the Pollux cask. The center space contains a canister filled with compacted fuel assembly hardware or consolidated rods from two PWR assemblies. Alternatively, unconsolidated spent fuel assemblies or chopped fuel rods could be placed in the casks. The Pollux cask system has a steel body (15Mn-6.3Ni steel, 15 cm thick), with a shielding overpack of 265-mm-thick ductile cast iron (GGG 40). The shielding overpack has two concentric rows of 75-mm-diameter holes that contain graphite rods for neutron moderation. The empty Pollux cask system weighs about 58 MT, and the loaded weight is about 65 MT. The system is 1.56-m outside diameter by 5.52-m long and will be backfilled with helium. The sealed cask has a screwed primary lid with elastomeric seals and a full-thickness welded secondary lid; the shielding overpack has only a screwed lid. Each Pollux cask can handle spent fuel that has been out-of-reactor for three years with a total heat generation rate of 20 kW (Einfeld and Popp 1986; Schneider et al. 1989, 1990, 1991; Spilker and Huggenberg 1991).

5.4.2 Spent Fuel Dry Storage Facility Designs and Experience

Dry storage has been developed, demonstrated, licensed, and used for small quantities of special fuels, but has not been widely used to date. Dry storage has been demonstrated at reactor sites and by loading LWR fuel into casks, followed by shipment to and storage at R&D sites. The dry storage in casks at AFRs has been deferred for a number of reasons, including a lack of current need (Schneider et al. 1990; NF 3/4/91). Results (described in Section 5.5) of dry storage demonstrations of casks and the spent fuel behavior confirm the maturity of this technology. Further cask development will concentrate on optimization, to be carried out by the cask suppliers (Einfeld and Popp 1986).

The AFR storage facilities at Gorleben and Ahaus are essentially identical, each comprising a large reinforced concrete hall (about 200 m long, 38 m wide, and 20 m high); the storage capacity of each is 1,500 MT in 420 cask storage positions, stored vertically. The Gorleben facility is licensed to store LWR spent fuel, vitrified HLW, and LLW/ILW. The Ahaus facility is licensed to store LWR and pebble bed reactor spent fuel. The AFR facilities are designed to provide natural convective air cooling of the dry storage casks (Muller 1986; Schneider 1990).

Dry storage of HTGR spent fuels has been conducted in facilities at KFA (AVR reactor fuel) and at the AVR (buffer storage) and THTR reactors. A dry storage facility for AVR reactor spent fuel, contained in a hot cell and using natural convection air cooling, became operational in 1981 at KFA (Julich Research Center). It has a capacity of 112 canisters, 950 fuel elements per canister, i.e., 106,000 elements total, which is the total load of the AVR core (0.62 MT). The canisters are filled with a helium-air mixture. The maximum canister center temperature is 40°C in the KFA dry storage facility (170°C at the AVR reactor buffer dry storage area). Measurements have been taken at the AVR dry storage facility from its startup, and there have been no problems. No deterioration of the fuel microspheres was found during experiments for any of the storage conditions (IAEA 1988; Kirch 1990).

At-reactor dry storage for spent THTR fuel has been used since 1988. THTR spent fuel is contained in canisters (2100 fuel elements per canister) that are stored in holes in concrete blocks with forced air cooling. The THTR core consists of 675,000 fuel elements (0.3 MTU, 6.6 MTTh). The canisters are filled with a helium-air mixture. The maximum canister center temperature during dry storage is 320°C at the THTR facility. Transport/storage casks (similar to those for LWR fuel) have been selected for further storage of both AVR and THTR spent fuel at the Ahaus AFR (Kirch 1990; IAEA 1988; Heimlich 1991).

A bunker or vault-type concept for spent fuel dry storage (named FUELSTOR) is being developed by Siemens/KWU (formerly Advanced Nuclear Fuels Corp.). The concept consists of a concrete vault that holds sealed stainless steel canisters containing spent fuel and helium gas that are stored horizontally in specially-designed racks. Cooling is by natural convection with ventilation inlets and outlets in the side-walls of the building. The facility can be designed to accommodate from 500 to 5000 MTU. The typical dimensions of a facility for AR storage are 100 ft long by 55 ft wide by 70 ft high, with a capacity for 500 MTU; the dimensions for an AFR facility are 180 ft long by 115 ft wide by 70 ft high, with a capacity for 3,000 MTU (Peehs et al. 1987; Valentine 1988; Johnson 1989).

5.5 R&D IN DRY SPENT FUEL STORAGE INTEGRITY

Research and development (R&D) work has been carried out in the FRG since 1979 to assess the integrity of LWR spent fuel assemblies and their ability to be handled throughout dry storage periods. The work has included full-scale dry storage cask demonstrations since 1982 (various GNS and TN casks, including Castor Ia, Ib, Ic, IIa, KRB-MOX casks, and TN-1300 casks) with irradiated PWR and BWR fuel elements (DOE 1989). The research programs have included theoretical predictions, laboratory work, and performance testing using actual LWR spent fuel. A comprehensive R&D effort was performed on LWR spent fuel dry storage behavior, for intact fuel and fuel with cladding defects. The program covered generic investigations, defect mechanism assessments, defective fuel rod behavior investigations, safety-related investigations, and verification experiments.

The tests have involved approximately 3,000 fuel rods, with maximum fuel cladding temperatures ranging from 250 to 430°C. The storage periods have ranged from one week to more than two years. During the tests, parameters relevant to spent fuel behavior, such as axial and radial cladding temperatures and cover gas radioactivity, were monitored. Post-test fuel inspections were carried out and compared with the pre-test data. Comparisons of the experimental and theoretical data on the cladding behavior and radioactivity release rates were performed to improve theoretical models and confirm long-term assertions.

There has been no evidence that any rods have failed in the FRG during dry storage in inert atmospheres. Dry storage under cask conditions with maximum temperatures in the range of 400°C is not expected to cause significant creep deformation (i.e., above 1%), cladding oxidation, or conditions that induce stress corrosion cracking or flaw propagation. Overall results indicated that dry storage in an inert atmosphere, following wet storage for about one year, is safe and reliable. The results of the R&D activities show that:

- The experimental findings and the theoretical analyses were in agreement.
- Measurements of tangential strain and cladding oxidation were near the detectability limit (<4 μm) under expected dry cask storage conditions in the FRG.
- Post-reactor creep rates were less than or equal to those of unirradiated Zircaloy. The primary creep depended on the stress state prior to shutdown.
- There were no indications of iodine stress corrosion cracking or crack growth in the Zircaloy cladding.
- The fission product release from spent UO_2 was low to negligible. Fission product releases occurred only in the heat-up phase and were very small. The release rates became zero after a few days under storage conditions.
- Tritium was distributed between the fuel (63%) and cladding (37%). Its release can be calculated by Sieverts' Law.
- Oxide and crud layers were adherent and did not spall from the fuel rod surfaces during dry storage.
- Minimal contamination of the cask inner surface and basket occurred.
- Fuel rods with cladding defects could be dried easily by the standard cask drying procedures. Water logging was not observed. Residual moisture of the cask atmosphere can be minimized.

- No defects caused by storage in a dry and inert atmosphere were observed even if storage conditions were beyond the design criteria. There were no indications of any further propagation of cladding defects under inert atmosphere dry storage conditions.

- The upper temperature limit considered to be safe and reliable is 420°C for dry storage in an inert atmosphere. Somewhat higher temperatures also appear to be acceptable.

5.6 PROJECTIONS OF LONGEVITY OF SPENT FUEL IN DRY STORAGE IN THE FEDERAL REPUBLIC OF GERMANY

Dry storage of Zircaloy-clad PWR and BWR fuel in different types of metal casks (Castor and TN series) has been demonstrated without causing any significant creep deformation, cladding oxidation, or conditions that induce stress corrosion cracking or flaw propagation. No rod failure due to dry storage has been observed during storage periods of about two years. These results are considered by the Germans to provide assurance for extended dry storage in inert cover gases at cladding temperatures up to about 420°C (Peehs and Rudolf 1991). It was demonstrated that creep strain comes to a halt after a few months. "It is therefore of no consequence how long the period of dry storage lasts, whether 40 years or 100" (Johnson 1989). The Pollux cask has been designed to keep the maximum creep to less than 1% at no more than 390°C for a period of 40 years, with a conservative cladding damage of 1% (Spilker and Huggenberg 1991).

If the cover gas during storage is assumed to be an oxidizing gas rather than an inert one, the storage temperature must be reconsidered due to cladding corrosion effects. After about three years, when the dry-stored cladding temperature falls below 250°C, cladding oxidation comes to a virtual halt; the annual increase in the oxide layer thickness at 250°C is less than 0.1 μm. "The length of the period of dry storage is thus of no importance as regards cladding corrosion either" (Johnson 1989). However, the oxidation of fuel in defective fuel rods necessitates a considerable reduction in the permissible initial storage temperature for an oxidizing storage environment. This process does not cease soon after storage begins. Therefore, investigations of UO_2 oxidation for storage periods of 40 and 100 years appear to be necessary (Johnson 1989).

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6.0 DRY STORAGE EXPERIENCE IN FRANCE

6.1 NUCLEAR POWER, REACTORS, AND SPENT FUEL

France's entry into commercial production of electricity using nuclear energy began in 1956 when the Marcoule G1, a 2-MWe natural uranium, gas-cooled, graphite-moderated reactor, was connected to the electrical grid. Since then, France has implemented a rapid growth of nuclear power. France leads the world's countries in the total fraction of electricity that is generated by nuclear power. (In 1989, 74.6% of France's electricity was generated by nuclear power.) At the end of 1990, France had 56 nuclear power reactors in operation with a gross electrical generation capacity of 57.8 GWe. Two of the operating nuclear reactors are gas-cooled reactors (GCR) with capacities of 465 and 555 MWe each (startups in 1971 to 1972), 52 are pressurized water reactors (PWR, startups in 1967 to 1990) with capacities ranging from 320 to 1380 MWe each, and two are fast breeder reactors (FBR, startups in 1974 and 1986) with capacities of 250 and 1240 MWe. A total of seven of the original gas-cooled reactors (startups in 1956 to 1972) and the only gas-cooled heavy-water reactor (startup in 1968) have been shut down. Six PWRs with a total capacity of 8.6 GWe are under construction, and five more PWRs are planned, with a total capacity of 7.4 GWe. One FBR has been indefinitely deferred or cancelled (NW 1991; NEI 1991).

The French government has been highly in favor of nuclear energy to provide the bulk of its electricity, particularly since the oil crisis in 1973. However, political and public pressure to re-evaluate the nuclear power program is growing, with concerns about safety, waste management, and facility decommissioning. Although France's plans for near-future addition of new nuclear power plants has slowed down, its planned addition of nuclear energy is third only to that of the former Soviet Union and Japan (NEI 1991; Schneider et al. 1991).

A fuel assembly in the French GCRs is similar to those in the U.K.'s Magnox reactors. An assembly generally consists of one large magnesium or magnesium-zirconium alloy-clad rod (cladding is 1.5 to 2 mm thick; some rods have a longitudinal hole in the center) filled with 0.8 to 25 kg of natural uranium metal. Fuel assemblies are inserted 15 to a channel in the core. Except for the Bugey-1 reactor with 12,780 assemblies, each reactor contains about 44,000 assemblies representing 350 to 450 MTU. Total burnup is 3000 to 3500 MWd/MTU. The reactors use carbon dioxide coolant and graphite blocks for moderation (NEI 1991).

Fuel from the Phenix fast reactor contains mixed plutonium and uranium oxides with an equivalent enrichment of 27.1%. The fuel is contained in 103 hexagonal assemblies, each having 217 pins, with an outside diameter of 6.6 mm, and a height of 17.93 mm. The fuel cladding is 0.45 mm thick and is Type 316 stainless steel; the Phenix core contains 4.3 MTHM (metric tons of heavy metal) and the nominal burnup is 100,000 MWd/MTHM (NEI 1990).

Fuel from the Superphenix fast reactor contains mixed plutonium and uranium oxides with an equivalent enrichment of 16.6%. The fuel is contained in 364 hexagonal assemblies, each having 271 pins with an outside diameter of 8.5 mm and a height of 2700 mm. The fuel cladding is 0.57 mm thick and is Type 316 stainless steel. The Superphenix core contains 31.5 MTHM and the nominal burnup is 44,000 MWd/MTHM (NEI 1990).

6.2 SPENT FUEL MANAGEMENT POLICY

The generation, transmission, and distribution of electricity in France is federally owned. Exceptions are the two fast reactors, which are co-owned by other entities, some of which are foreign (Leigh and Patridge 1991).

The nuclear power generator (Electricite de France, [EdF]) is responsible for interim storage of its spent nuclear fuel. All French nuclear power fuel has been and continues to be reprocessed in France (GCR and FBR fuels at Marcoule; GCR fuels and some FBR fuels until 1987 at La Hague; and PWR fuels at La Hague since 1976). Reprocessing is done to allow for future recycle of fissile materials in LWRs and FBRs. The high-level waste resulting from reprocessing has been and continues to be vitrified, and will eventually be disposed of in a French deep geological repository. France stores its spent fuel from PWRs and GCRs in the reactor water basins for about one year until it is shipped for reprocessing. Additional water pool storage for two to three years is provided for 10,000 MTU at the La Hague and for 800 MTU at the Marcoule reprocessing plants. These facilities are owned and operated by Cogema (another government corporation). Thus, France has had no need for extended storage of the vast majority of its spent nuclear fuel. The spent fuel from the two fast reactors is stored dry after an interim storage period in molten sodium. A dry storage facility has been constructed at Cadarache for storage of miscellaneous spent fuels (NEA 1986; Hammer 1988; Schneider et al. 1991).

6.3 STATUS OF SPENT FUEL STORAGE

Spent fuels from the nuclear power stations in France are stored in water pools for about one year until the fuels are sent to the reprocessing facilities, where they are stored in water pools while awaiting reprocessing. The water in pool storage of GCR fuels is kept at a pH of about 11.7 to minimize corrosion (Chometon 1990; Schneider et al. 1991).

A dry storage vault facility at Cadarache, the CASCAD facility, was started up in 1990 for up to 200 MTU of experimental fuels that would be impractical to reprocess (Bonnet and Guay 1991). In addition, a new fuel packaging and dry storage facility, called STAR, will start up in 1992. The storage system is part of the fuel active examination facility at Cadarache. The fuel will be packaged in hermetically-sealed containers that are dissolvable in the Marcoule reprocessing plant (NEW 1991).

A vault for dry storage of up to 9.5 MTU of spent FBR fuel has been used since 1985 at the Traitement Oxyde Rapide (TOR) reprocessing facility for FBR fuel at Marcoule. The facility is used for storing pre-canistered FBR fuel after 6 to 24 months from discharge from the reactors (IAEA 1988).

To provide the capability for dry storage in storage casks (in other countries currently and possibly for France in the future), Transnucleaire Company of France has developed and licensed two transportable storage casks for LWR spent fuel. The TN-24 cask has the capability to store 28 PWR fuel assemblies that are five years out-of-reactor or 32 PWR assemblies that are 10 years out-of-reactor. The casks are made of forged steel with walls approximately 22 cm thick. A layer of borated resin for neutron shielding is enclosed between the forged steel body and an external steel shell. Longitudinal copper fins transfer heat between the body and the smooth outer shell. A single bolted lid with two metallic gaskets provides a seal for transportation. For storage and for transportation following storage, a second, thinner forged steel lid with double gaskets is mounted over the primary lid. The cask and the space between the lids are filled with helium before sealing. The overall dimensions with removable impact limiters on each end for transportation are 2.65 m in diameter by 5.8 m long. The cask weighs up to 114 MT with a maximum payload of 15 MTU (Kirchner and Nagahama 1991).

6.4 EXPERIENCE IN DRY SPENT FUEL STORAGE

The CASCAD dry storage facility at the Cadarache Nuclear Research Center in France started up in 1990. This facility was built to store spent fuels from a wide variety of experimental spent fuels in France that would be impractical to reprocess. The facility stores fuels such as those from France's experimental gas-cooled heavy water reactor, with burnups of 15,000 to 20,000 MWd/MTU. The storage facility was built to provide extended storage for up to 50 years until the ultimate disposition of the fuels has been determined. The CASCAD facility is a concrete vault (with 850 m^2 of total floor area, 240 m^2 for fuel storage) cooled by air using natural convection (normally unfiltered exhaust) that flows across the storage tubes. The vault is partially below grade, and is below an overlying fuel handling cell. The spent fuel in helium-filled canisters is stored in a series of 319 vertical storage tubes that are sealed with a double-gasketed shield plug after loading with spent fuel. The maximum temperature of spent fuel cladding inside the spent fuel canisters is 125 to 180°C (115°C for fuels from heavy water reactors), depending on the fuel. Each of the stainless steel storage tubes (346 mm in diameter and 7 m high), which contain air around the spent fuel canisters, is rated to contain fuel with 600 W of heat. Barriers to prevent the escape of radionuclides are the spent fuel cladding, the canister around the spent fuel, and the sealed storage tubes. The vault, which can handle 150 to 200 MTU of special fuels, has a total heat removal capability of

200 kW, and is designed for possible future additions if necessary (Teulon 1986; IAEA 1988; Bonnet and Guay 1991).

The dry storage vault for spent FBR fuel at the TOR reprocessing facility at Marcoule consists of 77 tubes (300 mm in diameter by 9.85 m long, with spacer ribs to center the spent fuel canisters) encased in concrete in a vault with an overlying handling area. Each tube can hold spent fuel from the Phenix reactor in three canisters, and from the Super Phenix reactor in five storage canisters, stacked end to end. Each canister, with a diameter of 103 mm, holds one assembly of 93 fuel pins. A heat release rate of up to 4.3 kW can be handled in a tube, with a maximum fuel cladding temperature of 640°C. Heat removal is by forced-air ventilation through each tube (Sugier et al. 1982; Blomeke 1988; IAEA 1988).

A dry vault design has been developed for dry storage of LWR fuels for potential use at the La Hague reprocessing facility, but implementation has not started because its need has not yet been confirmed. The concept involves natural air convection cooling, with spent fuel placed in steel canisters with an inert atmosphere. About five years of storage in water will be required before spent fuel could be stored in such a facility (IAEA 1988).

6.5 R&D IN DRY SPENT FUEL STORAGE INTEGRITY

The French have been developing hardware for consolidation of LWR fuel rods in a dry environment and in water pools (Maillet 1989). No evidence was found of R&D in dry spent fuel storage technology, other than heat transfer developments for their two small dry storage facilities for special fuels.

6.6 PROJECTIONS OF LONGEVITY OF SPENT FUEL IN DRY STORAGE IN FRANCE

The CASCAD facility is claimed to allow "safe, economical interim storage of irradiated fuels for a period of 50 years" (Bonnet and Guay 1991). No other information on predictions of the longevity of dry-stored spent fuel could be found for France.

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7.0 DRY STORAGE EXPERIENCE IN INDIA

7.1 NUCLEAR POWER, REACTORS, AND SPENT FUEL

India started commercial nuclear power production in 1969 with its Tarapur 1 & 2 reactors (160 MWe each). It presently has eight nuclear power reactors in operation with a gross electrical generation capacity of 1,565 MWe. The operating nuclear power reactors in India currently include two boiling water reactors (BWRs), five pressurized heavy water reactors (PHWRs), and one small (13 MWe) fast breeder reactor (FBR). Except for the FBR, the capacities of the individual reactors vary from 160 to 235 gross MWe. The first two PHWRs (CANDU-type) were supplied by Canada; India has since developed an indigenous capability for reactor construction. A total of seven PHWRs are presently under construction, and twelve more nuclear power plants are planned, including ten PHWRs and two PWRs. By the year 2000, India plans to reach a target nuclear generation capacity of 10,000 MWe (Rao 1990; NEI 1991).

India is placing heavy dependence on nuclear power to augment its electric power generating capacity. It is planning a three-phase program: in the first phase, reactors fueled with natural uranium will be used; in the second phase, FBRs fueled with plutonium produced by first-phase reactors will be used, with breeding of ^{233}U in thorium blankets; in the third phase, thorium/^{233}U breeder reactors will be used. The fuel cycle strategy is designed to take advantage of the large thorium resources in India. India's overall plan is to achieve self-sufficiency in the entire fuel cycle (Schneider et al. 1991; Nair and Prasad 1991).

The majority of spent fuel in India is of the CANDU type with Zircaloy cladding and lower burnups than typical LWR spent fuel (see Section 4, "Dry Storage Experience in Canada," for description). A significant amount of BWR spent fuel and a small amount of FBR spent fuel also exists. The BWR spent fuel is clad in Zircaloy and is arranged in a 6 x 6 array (Rao 1990).

7.2 SPENT FUEL MANAGEMENT POLICY

India plans to reprocess spent fuels for recovery and reuse of the nuclear materials. It primarily uses at-reactor (AR) pool storage prior to reprocessing of the spent fuels. The schedule for construction of reprocessing plants will follow the plutonium demand for FBRs. This may involve storage of considerable quantities of spent fuel for varying periods (Rao 1990). India previously planned to reprocess spent fuel after a 100-day cooling period. Due to delays in planned availability of domestic reprocessing capability, India has increased its pool storage capacities at the reactors by using high-density racks. It is currently using four dry storage casks that were indigenously designed and built to supplement its pool storage at Tarapur. India is also constructing a pool-type, away-from-reactor (AFR) facility near Tarapur. Dry vault storage has also been studied, but pool storage technology was selected for AR-capacity expansion, at least at this time, due to India's greater level of experience with pool storage. India is, however, considering dry storage in concrete casks (Johnson 1989; Schneider et al. 1991; Srinivasan 1991).

India transports spent fuels to domestic reprocessing facilities using indigenously designed and built transport casks. It plans to vitrify the high-level wastes (HLW) resulting from reprocessing and dispose of them in a deep geologic repository. Interim storage of vitrified HLW is performed in a natural air-cooled facility at Tarapur (Harmon and Johnson 1984). Due to an inadequate transportation network, smaller reprocessing plants at several locations with co-located vitrification and vitrified HLW storage facilities are being implemented rather than a centrally-located, large-capacity reprocessing plant (Sethna et al. 1983; Schneider et al. 1991)

7.3 STATUS OF SPENT FUEL STORAGE

India uses the reactor pools to store the majority of its spent fuel. A water pool AFR storage facility with a capacity for 3312 BWR spent fuel assemblies (about 460 MTU) was commissioned in late 1990 near (0.5 km) the Tarapur Atomic Power Station (TAPS), with dual General Electric-supplied BWRs. This AFR is to supplement the AR pool storage at TAPS, and its capacity will accommodate the entire amount of spent fuel to be discharged from the two TAPS reactors during their expected 30 years of operation. The pool is 9 m wide by 13 m long by 13 m deep and is lined with stainless steel (Johnson 1989; Srinivasan 1991). Experience over the last two decades has shown that there is no deterioration of Zircaloy-clad fuel in pool storage (Rao 1990).

PHWR fuels are stored in AR pool facilities in horizontal stainless steel racks in relatively shallow pools at the Rajasthan and Madras Atomic Power Stations (RAPS and MAPS). The capacity of the spent fuel pool at each of the PHWR stations is equivalent to 10 years of spent fuel discharged from these reactors (Rao 1990). Shortage of wet pool storage space has necessitated rearrangement of the spent fuel racks at RAPS. The racks are now stacked up to 30 high to give a total capacity of 32,200 PHWR fuel assemblies (about 620 MTU). The provision exists to store RAPS spent fuel at the AFR facility at Tarapur in case of an emergency (Johnson 1989; Schneider et al. 1991).

Dry storage casks (described in Section 7.4), indigenously designed and built, are used to store a limited amount of BWR spent fuel at the TAPS reactor site. Dry vaults were studied in 1984 as a possible means to expand the limited AR spent fuel storage capacity, but India has generally decided to use proven pool storage technology. Preliminary design has been undertaken for using concrete dry storage casks at Tarapur. The conceptual facility would use concrete casks that would contain 230 to 240 PHWR assemblies each. The casks would set on a concrete pad at grade level. The facility could place 3600 fuel assemblies/yr into storage (Johnson 1989; Srinivasan 1991).

7.4 EXPERIENCE IN DRY SPENT FUEL STORAGE

Four dry storage casks have been used at TAPS since 1987 to store spent BWR fuel in an air atmosphere. Each cask is capable of storing 37 BWR fuel bundles (or 5.1 MTU). The design and fabrication of these casks was done indigenously. The casks are shielded with 120 mm of lead, a 40 mm steel outer shell, and a 12 mm stainless steel inner liner. A layer of stainless steel (6 mm thick) is used on the exterior surface for ease of decontamination. Interior stainless steel baskets are used to support the fuel bundles. The cask cavity is about 1.2 m in diameter and about 4.3 m long. The overall dimensions of the casks are 1.5 m in diameter by 4.7 m in length. Each cask weighs approximately 55 MT. Fuel assemblies having a burnup greater than 13,000 MWd/MT and a minimum cooling period of 10 years in water are stored in these casks (Johnson 1989; Rao 1990; Schneider et al. 1991).

Closure of the dry storage casks is done with two separate bolted-on covers, one internal shielding cover and an outer sealing cover. Spiral-wound asbestos-filled metallic gaskets are used in the covers to ensure good sealing against leakage. The exterior surface of the casks has not exceeded 75°C. The dry storage casks are designed to withstand the 9-m drop test condition and the punch test requirement for transportation usage (Rao 1990).

The storage casks will also be used to transport spent fuel assemblies from the Tarapur reactors to the wet AFR storage facility. Only fuel with a burnup of greater than 13,000 MWd/MT and a minimum cooling period of 10 years will be shipped to the wet AFR storage facility, due to the dry cask design limits (Rao 1990).

The fuel in the dry storage casks is to be unloaded into a pool for examination to determine the effect of dry storage in air on the spent fuel (Srinivasan 1991).

7.5 R&D IN DRY SPENT FUEL STORAGE INTEGRITY

A heat transfer mathematical model study of a spent fuel dry cask is in progress. The model covers the heat generation due to radioactive decay, conductive heat transfer in the cladding, natural and mixed convective heat transfer to a gaseous coolant environment, and heat transfer from the gaseous coolant to the outer container. Experiments are also proposed to be conducted on a scaled laboratory model to validate the theoretical results. This study will facilitate a reliable thermal design for dry storage of spent fuel in casks (Rao 1990).

7.6 PROJECTIONS OF LONGEVITY OF SPENT FUEL IN DRY STORAGE IN INDIA

No information on projections of longevity of dry-stored spent fuel could be found for India.

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8.0 DRY STORAGE EXPERIENCE IN JAPAN

8.1 NUCLEAR POWER, REACTORS, AND SPENT FUEL

Japan started nuclear power production in 1965 at its Tokai Japan Power Demonstration Reactor (JPDR) (13 MWe), followed by startup of its first industrial-scale reactor in 1966, the Tokai Japco (166 MWe) reactor. Japan presently has 41 nuclear power reactors in operation with a gross electrical generation capacity of 31,645 MWe. The Japanese operating nuclear reactors include 21 boiling water reactors (BWRs), 17 pressurized water reactors (PWRs), one Magnox reactor, one fast breeder reactor (FBR), and one light water-cooled, heavy water-moderated reactor (LWCHWR, an advanced thermal reactor, or ATR, at Fugen). The capacities of the individual light water reactors (LWRs) vary from 357 to 1175 gross MWe. One reactor, the Tokai JPDR, has been shut down (NEI 1991).

Five BWRs, six PWRs, and one FBR are presently under construction in Japan, and an aggressive planning program for future nuclear reactor construction is being pursued. In the year 2000, the estimated nuclear generation capacity is estimated to be approximately 50 GWe. The Japanese government actively promotes nuclear power as the primary means of reducing dependence upon foreign energy sources. The Japanese strategy is to install LWRs for near-term power production, develop ATRs and FBRs, and eventually depend heavily on fusion power (Harmon and Johnson 1984; NEI 1991; Kawasaki and Ogata 1991).

The spent fuel in Japan consists primarily of LWR fuels, both BWR and PWR spent fuels, but small amounts of FBR, Magnox, and LWCHWR spent fuels also exist. The BWRs typically use 8 x 8 pin arrays and the PWRs use 17 x 17 pin arrays. The average enrichment of the BWR fuel is 3% and the PWR fuel is 3.2% (Schneider et al. 1991).

8.2 SPENT FUEL MANAGEMENT POLICY

The spent fuel management policy in Japan is to store the spent fuel in the reactor pools for a period of two to three years, followed by shipment offsite for reprocessing. All spent fuel generated in Japan is planned to be reprocessed, and the uranium and plutonium recovered is planned to be recycled for future power generation, to breeder reactors in the long term and thermal reactors (ATRs and LWRs) in the near term. Presently, Japanese spent fuel is being reprocessed abroad (in France and the United Kingdom), while domestic commercial reprocessing capability is being developed at a new fuel cycle center at Rokkasho-mura (800 MTHM/yr), expected to be operational about 1997. Domestic reprocessing is conducted at a smaller scale (210 MTHM/yr) at the Tokai reprocessing plant (Watanabe et al. 1988; Schneider et al. 1991).

Spent fuel is currently stored in water pools, but research on dry storage technology is being supported to provide more flexibility for future spent fuel management (Watanabe et al. 1988; Shiomi et al. 1992).

8.3 STATUS OF SPENT FUEL STORAGE

Spent fuel storage is carried out in the pools at the reactors and at the Tokai reprocessing plant. Approximately 2250 MTU of spent fuel was being stored in at-reactor pools as of March 1990; about 5000 MTU of spent fuel has been shipped to domestic and foreign reprocessing facilities. The Tokai reprocessing plant has spent fuel storage pool capacity of 97 MTU, and the planned Rokkasho-mura reprocessing plant will have a pool storage capacity of 3,000 MTU (Kawasaki and Ogata 1991; Schneider et al. 1991). Sufficient wet storage experience has been accumulated in Japan to ensure its reliability and safety in the long term (Watanabe et al. 1990).

A dry storage facility has been in operation since 1982 at the Japanese Atomic Energy Research Institute (JAERI) and has a capacity for 30 MTU of Japan Research Reactor No. 3 (JRR-3) uranium metal fuel. This is the first dry storage facility for spent fuels in Japan (IAEA 1988; Shirai et al. 1991).

8.4 EXPERIENCE IN DRY SPENT FUEL STORAGE

A drywell vault facility (30 MTU capacity) has been in operation since 1982 and presently is used to store about 15 MTU of JRR-3 research reactor fuel. The JRR-3 fuel consists of natural uranium metal rods (25 mm diameter) clad in aluminum alloy tubes (950 mm long). The fuel burnup is 800 MWd/MTU with a decay heat of 0.5 W. The fuel was cooled for a minimum of about seven years (2500 days) in the at-reactor pool prior to dry storage (IAEA 1988; Shirai et al. 1991).

The JRR-3 dry storage facility consists of a concrete building 12 m long by 13 m wide by 5 m deep and is below ground level. The spent fuel elements are contained in helium-filled stainless steel canisters (with 0.5 cm thick walls), which are welded shut. Each canister can accommodate 36 fuel elements. The fuel was dried for 1 week in a hot cell before seal welding into canisters. The canisters are emplaced in drywells within stainless steel liners (each 36 cm in diameter, 2.5 m deep, and 0.8 cm thick) and covered with a carbon steel shield plug. There are a total of 100 drywells in the concrete facility. Forced air is circulated in an annular space between the concrete and canister. The interior of the drywell is kept at a negative pressure. There have been no abnormal occurrences or requirements for remedial action. About 18,000 fuel elements or 15 MTU (half of the 3,600-fuel element, 30-MT capacity) are presently stored in the facility (IAEA 1988; Shirai et al. 1991).

In 1987 (after five years of dry storage), two canisters from the JRR-3 dry storage facility, containing a total of 72 spent fuel elements, were examined in the hot laboratory at JAERI. The visual inspection of the spent fuel elements and the canisters revealed no defects such as cracks or corrosion. No leaky canisters or spent fuel elements were detected by helium leak testing and gamma-ray measurements. From the x-ray radiography, no abnormalities were observed for any of the spent fuel elements (IAEA 1988; Shirai et al. 1991).

Japanese Zircaloy-clad fuel has been kept in dry transportation casks during sea transport for as long as 3.7 months, at estimated cladding temperatures as high as 385°C. Each cask holds 12 PWR assemblies and each ship carries 10 to 20 casks. Cask atmospheres have been inert in some cases, air in others. Several hundred assemblies have been shipped from Japan to Europe in this way without any evident cladding failure due to handling and shipping, as of 1984 (Harmon and Johnson 1984).

The Central Research Institute of the Electric Power Industry (CRIEPI) has tested a 110-MT-class Japanese transport/storage cask, which is analogous to a Castor V/21, manufactured by GNS in Germany. It is the fruit of a cooperative agreement between GNS, Mitsubishi Metal Corp., and Mitsui and was manufactured by Nippon Steel and Kubota. Tests have been carried out on heat transfer, lid sealing, cask dropping, materials performance, toppling of vertically-oriented casks, and creep deformation on Zircaloy fuel cladding (NW 1988; Shimoi et al. 1992).

A concept for storing vitrified high-level wastes (HLW) in passively-cooled dry vaults has been developed for a site in northern Japan. The storage vault area is a combination steel-frame structure with metal sides and a modular, below-ground concrete structure. The vault can accommodate up to 2500 canisters, which are stored in sealed sleeves that each hold five stacked canisters. The areas containing the canisters are kept at slightly lower pressures than surrounding areas to prevent the spread of any potential airborne contamination. The canisters are cooled by natural convection air. The concept was determined to be technically and economically feasible and applicable to storage of spent nuclear fuels. It is expected that the facility will be able to store wastes for up to 50 years (Miller et al. 1991).

8.5 R&D IN DRY SPENT FUEL STORAGE INTEGRITY

The Japanese have performed oxidation tests (in air and argon-air mixtures) on Zircaloy cladding and UO_2 pellets and fuel rods (with and without irradiation). In the oxidation tests of artificially-defected fuel rods, the damage rate of the fuel rod was a function of temperature, atmosphere, size of the defect, and previous irradiation history. Damage behavior of rods in low oxygen partial pressures was quite different from that in air (Kawasaki and Nakamura 1991).

From the oxidation test experimental results, the allowable dry storage temperature for intact spent fuel rods in air was estimated by using the following assumptions: 1) temperature is a constant, 2) cladding thickness is 0.57 mm (PWR 17 x 17 rod), 3) air storage period is 30 years, and 4) the acceptable amount of oxidation during storage is 10% of the cladding thickness. With these assumptions, the upper limit of the temperature for dry storage of intact fuel rods in air was calculated as 328°C, for a storage period of 30 years (Kawasaki and Nakamura 1991).

For fuel rods with defects, the rate of oxidation of UO_2 is much higher than that of Zircaloy, and the resulting volume increase damages the fuel rod. With the assumption that the acceptable amount of circumferential swelling is 1%, the upper limit of the dry storage temperature was calculated to be about 160°C (for defective fuel in air) for a storage period of 30 years. In low oxygen partial pressures, circumferential deformation was not detected in any irradiated specimens. The damage speed in low oxygen pressures is expected to be considerably slower than that in air. Therefore, the allowable storage temperature in a low oxygen pressure environment is estimated to be markedly higher than that in air (Kawasaki and Nakamura 1991).

Creep tests of BWR (completed) and PWR spent fuel cladding are being carried out at CRIEPI to establish the allowable temperatures for dry storage of spent fuel in an inert gas atmosphere. Maximum temperatures to be expected for dry storage of spent fuel in cast iron casks, having a maximum external temperature of 82°C, were calculated using the Department of Transportation (DOT) code. For PWR spent fuel (17 x 17 type) five years out-of-reactor, with a burnup of 48,000 MWd/MTU, the maximum temperature to be experienced by a centrally located fuel rod was calculated to be 361°C in a helium atmosphere and 466°C in an argon atmosphere. In the case of BWR spent fuel with a lower burnup of 39,000 MWd/MTU, the maximum temperature to be experienced was calculated to be 205°C (Tanaka 1985; Watanabe et al. 1990; Mayuzumi 1990; Kawasaki and Ogata 1991). Proposed maximum cladding temperature limits due to creep were plotted as a function of internal fuel pressure (at 25°C) and cooling time (1 to 5 years). Example proposed limits for PWR cladding are, at an internal pressure of 10 atmospheres and after 5 years cooling, about 415°C; at internal pressure of 50 atmospheres and after 5 years cooling, about 350°C; and after 1 year cooling, about 400°C. Example proposed limits for BWR fuels for those same conditions are about 450°C, 410°C, and 430°C, respectively (Saegusa 1991b).

Various spent fuel storage technologies have been evaluated by the Japanese, primarily on an economic basis. Dry casks, vaults, and pool storage of spent fuel in various capacity facilities were compared (Saegusa 1991a).

8.6 PROJECTIONS OF LONGEVITY OF SPENT FUEL IN DRY STORAGE IN JAPAN

From the experience gained during eight years of dry storage of uranium metal/aluminum alloy-clad JRR-3 fuel in helium, it was concluded that "spent fuels in the dry storage facility will be stored without any deterioration for a very long time" (Shirai et al. 1991).

For a storage period of 30 years, the upper limit of the temperature for dry storage of intact LWR fuel rods in air was calculated as 328°C (to allow for oxidation of 10% of the cladding). For a storage period of 30 years, for defective fuel cladding in air, the upper limit of the dry storage temperature was calculated to be about 160°C (Kawasaki and Nakamura 1991).

In tests of the durability of metallic seals used in dry spent fuel storage casks, it was determined that a metallic seal should maintain its negligible leakage rate for about 50 years (Saegusa 1991b).

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9.0 DRY STORAGE EXPERIENCE IN SWITZERLAND

9.1 NUCLEAR POWER, REACTORS, AND SPENT FUEL

Switzerland has had nuclear power since starting up its first pressurized water reactor (PWR) in 1969. There are now five operating nuclear power plants with a total installed capacity of about 3000 MWe net, which provided 42.6% of the total Swiss electricity in 1990. Three of the nuclear reactors are PWRs (350 to 920 MWe design net capacity) and two are boiling water reactors (BWRs) (306 to 942 MWe design net capacity); the last one starting operation in 1984. One new 1000-MWe PWR and two unspecified nuclear power reactors (total capacity 2000 MWe) have been indefinitely deferred due to public opposition to nuclear power. A referendum vote in 1990 resulted in a 10-year moratorium on new nuclear reactor construction (Ospina 1990; NEI 6/1990).

The federal government and electric utilities are in favor of nuclear power to meet Switzerland's increasing demand for electricity, but local and public opposition has delayed its expansion. Some district heating with an existing nuclear power plant was under construction but was abandoned (Schneider et al. 1990; NEI 6/1990).

9.2 SPENT FUEL MANAGEMENT POLICY

In Switzerland, the nuclear electric utilities are responsible for spent fuel interim storage, transportation, and reprocessing. The National Cooperative for Storage of Radioactive Waste (NAGRA), which is a private organization comprised of the nuclear electric utilities and other users of radioactive materials, is responsible for the ultimate disposal of the radioactive wastes. The Swiss have selected a fuel cycle involving reprocessing and plutonium recycling; however, the option of direct disposal of spent fuel is also open. All Swiss spent nuclear fuel to be discharged through 1993 is contracted to be reprocessed in France and the U.K. (Schneider et al. 1990).

Spent fuel is typically stored in the spent fuel pools at each reactor site for three to twelve years, which is nominally the capacity of the pool storage at each reactor, until the spent fuel is shipped to foreign reprocessors. However, available pool storage capacity is diminishing, so dry storage in metal casks has been selected to provide any excess storage requirements. The nuclear electric utilities formed a company (Zwischenlager Würenlingen AG) to plan, construct, and operate a central interim storage facility (Schneider et al. 1990; Ospina 1990) for spent fuel stored dry in metal casks. This facility, ZWILAG, which will include storage of all radioactive wastes in Switzerland, is located adjacent to the Paul Scherrer Institute in Würenlingen.

9.3 STATUS OF SPENT FUEL STORAGE

When the Swiss decided they would need some interim storage capacity for spent nuclear fuel beyond that provided by reactor pools, they evaluated the storage options and decided to use dry storage in transportable storage casks. Their first and currently only use of dry storage is licensed cask storage (beginning in 1983) of consolidated fuel from the DIORIT reactor, a 30-MWth heavy water research reactor that was shut down in 1977. The DIORIT fuel was placed in one CASTOR-1c-DIORIT cask in 1983 (Ospina 1986). However, as of April 1991, all other spent fuel in Switzerland is stored in reactor pools, and there is no indication of other dry storage before the interim storage facility becomes operable in about 1995 (Ospina 1988; Ospina 1990).

The future central interim storage facility for spent nuclear fuel in Switzerland is designed for a total of 184 to 200 casks (including casks for storage of vitrified HLW), for storage of up to about 1550 MTU of spent fuel. This is sufficient storage capacity for spent fuel and vitrified HLW arisings until about year 2010, and can be expanded if necessary. The Castor-type casks will be stored inside a building (Ospina 1988; Ospina 1990).

9.4 EXPERIENCE IN DRY SPENT FUEL STORAGE

The only dry storage of spent nuclear fuel to date in Switzerland, the fuel from the DIORIT research reactor, consists of about 2.5 MTU of natural and low-enriched uranium consolidated under water into 42-mm-diameter calandria tubes, four rods in each tube. A total of 350 fuel assemblies (2.3 m in overall length, with four Zircaloy-2-clad rods in each assembly) were consolidated to about 50% of the original assembly volume and placed into a single dry storage cask (Castor-1c) at the DIORIT reactor site. The calandria tubes were placed in two vertical layers in the upright cask. The nodular cast iron Castor 1c cask's overall dimensions are 1.78 m in diameter by 5.56 m high; its cavity is 0.693 m in a square cross-section that is 4.47 m high. (See Section 5, "Dry Storage Experience in the Federal Republic of Germany," for more descriptive information on the Castor-1c cask.) (Ospina 1986; Ospina 1987; IAEA 1988).

The initial maximum helium cover gas temperature in the DIORIT-Castor storage cask was calculated to be 180°C, compared with the Swiss allowable temperature of 410°C. The final leak rate of the cask, which uses one elastomeric and one metallic seal on each lid was less than 1×10^{-7} mb-L/sec. After two years of cask monitoring, the cask was transported to another storage site and reoriented to the horizontal storage position. The initial two-year monitoring included measurements of external temperatures, radiation dose rates, and the pressure between the two cask lids, as well as periodic purging of the helium at six atmospheres between the two cask lids. There have been no unusual occurrences. No fuel has been retrieved for inspection (Ospina 1986; Ospina 1987; IAEA 1988).

9.5 R&D IN DRY SPENT FUEL STORAGE INTEGRITY

There is very little evidence of major R&D in Switzerland on dry spent fuel storage; the Swiss appear to be using R&D and experience information developed primarily in the Federal Republic of Germany.

As of 1984, the Swiss were developing a unique modular dry vault storage concept, called MODREX. In this concept, spent fuel would be placed in canisters and placed vertically in individual cavities in a concrete silo-like vault. The canisters would be cooled by natural convection air in an enclosed system in which the hot primary air coolant was cooled by heat pipes containing methanol, ammonia or water. This concept, however, was never fully developed (IAEA 1984).

9.6 PROJECTIONS OF LONGEVITY OF SPENT FUEL IN DRY STORAGE IN SWITZERLAND

No information on predictions of the longevity of dry-stored spent fuel could be found for Switzerland.

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10.0 DRY STORAGE EXPERIENCE IN THE UNITED KINGDOM

10.1 NUCLEAR POWER, REACTORS, AND SPENT FUEL

The United Kingdom (U.K.) has used nuclear power for electricity generation since starting up its first four Magnox reactors in 1956. The U.K. has 37 operating nuclear power plants with a total installed capacity of about 12,400 MWe net. These reactors, and several others shut down in 1989 to 1990, provided 20.8% of the total U.K. electricity in 1989. Twenty-two of the operating nuclear reactors are Magnox Gas-Cooled (carbon dioxide-cooled, graphite-moderated) Reactors (60 to 665 MWe design gross capacity, start-ups 1956 to 1972, 3 shut down in 1989 to 1990), 14 are Advanced Gas-Cooled (AGR, carbon dioxide-cooled, graphite-moderated) Reactors (660 MWe design gross capacity, started up during 1976 to 1989), one is a Steam-Generating Heavy Water Reactor (SGHWR, 100 MWe capacity, shut down in 1990), and one is a prototype Fast Breeder Reactor (PFR, 250 MWe capacity). The last reactor to be commissioned (an AGR) started operation in 1989. The only new reactor under construction is a 1250-MWe PWR planned to be completed in 1994 to 1995. Three PWRs and one 1500-MWe FBR were planned but have been indefinitely deferred (NEI 1990a; NEI 1990b).

The present U.K. government is in favor of nuclear energy to meet its continuing demand for electricity. However, public opposition has curtailed expansion of nuclear power and has resulted in a four-year moratorium, which will be reviewed in 1994, on constructing nuclear power stations. Projections for nuclear power in the U.K. are for reduced capacity in the next decade or so, due to shutdown of a number of old reactors without equivalent replacement. The strategy for future use of nuclear power will be reviewed in 1994 (NEI 1990a; NEI 1990b).

A fuel assembly in a Magnox reactor consists of a single rod with natural uranium metal fuel clad in a magnesium alloy (99.2% magnesium and 0.8% aluminum) that is 2.0 to 2.5 mm thick. Each rod varies from 74 to 110 cm long and from 2.9 to 5.0 cm in diameter (and typically has longitudinal fins), with the larger dimensions generally used in the newer reactors. The total burnup in the Magnox reactor fuel ranges from less than 4000 to 5400 MWd/MTU (NEI 1991a).

A fuel assembly in an AGR consists of UO_2 fuel contained in 36 rods made of stainless steel (20% chromium and 25% niobium) that is 0.37 to 0.4 mm thick and has numerous circumferential ribs. The fuel is enriched to 2.1 to 2.8% ^{235}U and is irradiated to 18,000 to 21,000 MWd/MTU. The fuel pins are fixed in a graphite sleeve with graphite grid spacers to form an assembly (Addison 1987; Sills 1989; NEI 1991).

The SGHWR at Winfrith used UO_2 pellet fuel in Zircaloy-clad fuel rods that were 12.2 mm in diameter (cladding thickness was 0.597 mm) by 3805 mm long. Fifty-seven fuel rods in a circular array formed each fuel assembly. The original fuel was enriched to 2.3% ^{235}U and irradiated to about 21,000 MWd/MTU (NEI 1990b).

10.2 SPENT FUEL MANAGEMENT POLICY

In the U.K., the generation, transmission and distribution of electricity by public utilities was federally owned from 1948 until March 1990. Except for the nuclear power plants, which are owned by four government corporations, the electrical industry was privatized in 1990 (NEI 1990b).

The nuclear power generators are responsible for interim storage and transport to fuel reprocessing of their spent fuel. Since 1952, the U.K. has reprocessed all of its Magnox spent fuel, and the resulting vitrified high-level waste will be disposed of eventually in a deep geologic repository. Reprocessing is expected to continue for Magnox fuels for Nuclear Electric, the U.K.'s largest nuclear power utility, at least until 2005, when its current fuel reprocessing contract expires. At that time, the possibility of dry storage will be re-evaluated (NEI 1991b). Scottish Nuclear, Ltd. reviewed its policies for handling spent fuel from its AGRs after 1995, which include the current wet storage followed by reprocessing, long-term dry storage followed by reprocessing, or long-term dry storage followed by direct disposal. Scottish Nuclear announced in February 1992 that more than 75% of the spent fuel to be discharged in the future will be dry-stored at its two nuclear power plant sites for up to 50 years in vaults (NW 1992). A total of 1320 MTU of the expected 4000 MTU to be discharged during the life of the existing AGRs is committed to reprocessing in the THORP reprocessing facility after its startup in 1992; the disposition of the remaining 2700 MTU is yet to be determined, but up to 2000 MTU may be placed in dry storage (NEI 1990b; Hinton 1990).

10.3 STATUS OF SPENT FUEL STORAGE

Spent fuels from the nuclear power stations in the U.K. are generally stored in water pools until the fuel is reprocessed. Because of relatively high corrosion rates of the Magnox fuel cladding while in water storage at neutral pH, the Magnox fuels are stored for about six to twelve months in alkaline water (pH about 11.5) before being transferred to dry storage or to reprocessing. The cladding of the AGR fuels is somewhat more resistant to water storage, but it also suffers from aqueous (intergranular) corrosion due to sensitizing during reactor operation, and its water storage is limited to about 10 to 15 years in alkaline water. Some failure of water-stored AGR fuel was encountered in the late 1970s and early 1980s, but some water-stored AGR fuel has also been satisfactorily stored for 15 years (Sills 1989; Robinson 1989; Hands and Addison 1989; NF 1990; NEI 1990b).

The reprocessing facilities at Sellafield had only minimal short-term water pool storage until 1962, when a 1500-MTU pool for short-term storage of Magnox fuel was constructed. Another short-term wet storage facility with three pools having a capacity of 2250 MTU for Magnox fuels was constructed in 1985. For improved control of Magnox fuel cladding corrosion, Magnox fuel stored at Sellafield has been placed in sealed containers in highly alkaline water (pH about 13) with low chloride and sulfate content at the Fuel Handling Plant starting in 1986. This has resulted in low corrosion rates (less than about 0.2 μm/year) with very low local corrosion noted in the open storage in pools with pH of 11.5 (Addison 1987; Sills 1989).

AGR fuel is stored in pools for less than three years at the reactor sites, after which it is transferred to the storage pools at the reprocessing facilities at Sellafield, a procedure followed since 1978. All AGR spent fuel discharged to date has been stored in water pools, awaiting reprocessing or other disposition. In 1981, a storage pool (using water at pH of 7) for 550 MTU was constructed at Sellafield. An additional capacity of 1680 MTU of pool storage (with a receiving and packaging facility) was commissioned in 1986. At this facility, the graphite content of the fuel assemblies is removed and the spent fuel rods are consolidated into one slotted can for each three assemblies (0.86 MTU). The cans of consolidated rods are stored in water pools (with pH of about 11.4 and low chloride content) until the fuel rods are reprocessed. Examination of some assemblies in 1989, after six months in the reactor pools and about 1.8 years in the new alkaline storage facility, revealed corrosive attack generally less than 10 μm, with only two regions of 65 and 100 μm. A second additional pool with storage capacity for 3200 MTU at Sellafield has recently been commissioned. (Addison 1987; Groom 1988; Sills 1989; Robinson 1989; Hands and Addison 1989; Hinton 1990).

Spent fuel from foreign LWRs to be reprocessed in the U.K. has been stored for short to moderate time periods in pools at Sellafield since the 1960s. The method of storage has progressively changed from open racks to individual metal "bottles" to "multi-element bottles." The latter are sealed cans that can fit into a transportation cask. Ten years of monitoring the condition of LWR fuels stored at Sellafield for periods up to 20 years has revealed no significant deterioration of the fuel rods or hardware (Addison 1987; Hands and Addison 1989).

The spent fuel from the shutdown SGHWR, which is Zircaloy-clad, has been stored in water.

10.4 EXPERIENCE IN DRY SPENT FUEL STORAGE

Dry storage of Magnox spent fuel was initiated in modular vaults (or cells) in 1972 at the Wylfa nuclear power station site. The first vault consisted of three modules (each comprised of a cylindrical tank that contains the storage tubes for the spent fuel), each with a storage capacity of 83 MTU. Fuel rods are moved directly (within hours after reactor discharge) from the two reactors to storage tubes in the modules within the common reactor building. The fuel assemblies in each cell are stacked end to end in 588 sealed vertical tubes (11 m long) arranged concentrically in 11 rings. The fuel is stored there for at least 150 days in a dry carbon dioxide atmosphere (with less than 0.1% air content) at 3 psig. Heat is transferred from the storage tubes to a natural passive thermosyphon air circulating system to keep the fuel cladding temperatures no higher than about 365°C. This vault system has been very reliable, and no degradation of the fuel had been detected as of 1991 (Harmon and Johnson 1984; Cundill et al. 1986; IAEA 1988).

Because of the small capacity of the original Wylfa modular vaults, two additional vaults were commissioned, with a capacity of 350 MTU each, in 1979 and 1980. These concrete vaults are used to store Magnox fuel from the Wylfa station in natural draft, dry-air cooling after the initial 150-day storage in carbon dioxide-filled storage tubes. The maximum temperature of the fuel cladding in these vaults is 150°C to minimize damage to the Magnox cladding and fuel. Each of the air-cooled storage vaults is a concrete box 60 m long, 11 m wide and 4.5 m high. Fuel is stored vertically in 150 skips on the floor; each skip is a matrix of sealed tubes in a 12 by 16 array. Heat is transferred through the tubes to the naturally circulating air coolant (Harmon and Johnson 1984; Cundill et al. 1986; IAEA 1988).

In July 1990, some Magnox fuel assemblies stored in the Wylfa dry storage facility were found to have badly corroded cladding. One assembly broke while it was being moved, and five others appeared to have similar corrosion damage (Observer 1990). A total of 46 fuel elements were eventually found to have some damage. The damage was from a leak of rain water dripping from a roof joint onto the fuel elements. The damaged fuel elements are to be placed in individual cans and removed for reprocessing (NF 1991).

Scottish Nuclear, Ltd. (the smaller of the two nuclear utilities in the U.K.) recently opted for long-term storage of its future AGR fuels in Wylfa-type, air-cooled vaults at its two nuclear power stations starting in 1995 to 1996. Up to 2000 MTU of spent fuel could be stored for up to 50 years. The spent fuel will be placed in sealed storage tubes (eight elements each) with 180 tubes in each air-cooled vault (which can store 62 MTU). Two three-vault facilities will be constructed initially with subsequent vaults constructed as needed (NW 1992).

Before 1988, the U.K. had considered dry storage vaults for interim storage of AGR fuels after about 100 days of in-pool storage at the reactors (and also for interim storage of PWR fuel). The concept under consideration was to store AGR assemblies in high-integrity steel canisters that were placed in air-cooled modular concrete vaults. Each canister would contain four AGR assemblies (or one PWR assembly) in an atmosphere of high-purity dry argon. Each vault would store 1250 containers stacked two high, or about 210 MTU of AGR fuel or 614 MTU of PWR fuel (IAEA 1988).

10.5 R&D IN DRY SPENT FUEL STORAGE INTEGRITY

After storage for four years in the dry carbon dioxide environment in storage tubes in the Wylfa modular vault, examination showed the Magnox spent fuel to be unaffected. Storage for five years in dry carbon dioxide or dry air has resulted in no observations of storage-related defects (with the exception of fuel damage by contact with leaking rain water in one vault, discussed in Section 10.4) (Harmon and Johnson 1984).

The U.K. has investigated the oxidation behavior of irradiated UO_2 fuel as a function of oxygen partial pressure in carbon dioxide and nitrogen, and temperature; the results have been reported in documents from the IAEA's BEFAST program. Spent fuel fragments from AGR fuel pins were heated individually in crucibles in controlled atmospheres with oxygen partial pressures from 0.005 to 3 vol% and temperatures 250 to 450°C at one atmosphere pressure. It was found that, at temperatures below 350°C and oxygen levels at 0.05% or greater, the time for oxidizing UO_2 to powder was reduced only by a factor of about four compared with air. These times, which vary with temperature, were about 100 hours at 260°C or about 30 years at 100°C. Thus, even small quantities of oxygen at 260°C result in a very high rate of oxidation. It has been found that in air storage, the relative humidity must be less than 30% to maintain the integrity of the mild steel tubes containing the spent fuel. Based on U.K. data, the U.K. limits on temperature of defected fuel (not cladding) exposed to air under abnormal conditions are 300°C in dry air and 250°C in saturated air, due to possible ignition of the UO_2. R&D on storage of AGR fuel in air and air-inert gas mixtures is in progress (IAEA 1988; Wood et al. 1991; Simpson 1991).

The U.K. is carrying out a number of design/development activities on specific aspects of dry storage, including air circulation and heat transfer (including outside weather effects), techniques for seal welding steel fuel canisters, instrumentation for on-line measurements of emissions from dry storage facilities, modeling of sensitization stainless steel cladding, and the potential for post-irradiation annealing (IAEA 1988).

10.6 PROJECTIONS OF LONGEVITY OF SPENT FUEL IN DRY STORAGE IN THE UNITED KINGDOM

The temperature limitation for storage of Magnox fuel in dry carbon dioxide is about 365°C and about 150°C in dry air. The U.K. limits on temperature of defected fuel (not cladding) exposed to air under abnormal conditions are 300°C in dry air and 250°C in saturated air, due to possible ignition of the UO_2. Storage-related defects of Magnox fuel with dry air or carbon dioxide have never been observed, other than the cladding failures at the Wylfa dry storage facility in 1990 caused by contact with leaking rain water. Based on the positive storage experience available for time periods up to about five years (after which the Magnox fuel is generally reprocessed), several decades of storage of Magnox fuel in air is considered to be possible (IAEA 1988; NF 1991).

R&D on storage of stainless steel-clad AGR fuel in air and air-inert gas mixtures is in progress, and some information is available. Final projections of storage of defected AGR fuels in air are not available, but extrapolation from U.K. data of time for AGR spent fuel to become a powder gives a temperature limit of about 100-150°C for 30 years. One report (Agarwal et al. 1989) states that the air-cooled vault concept under consideration in the U.K. could act as a "50-year buffer" between fuel discharged from the AGRs and the national fuel reprocessing facilities (IAEA 1988; Wood et al. 1991; Simpson 1991).

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11.0 DRY STORAGE ACTIVITIES IN OTHER COUNTRIES

The preceding sections presented information on dry storage of spent nuclear fuel in countries with the most active dry storage programs, either currently or in the past, or about which more information is available. This section provides brief highlights of dry storage activities for spent fuel in other countries with known current or past activities. Information on dry storage from these countries (Argentina, the former German Democratic Republic, Italy, South Korea, Spain, and the former USSR) is presented in the subsequent subsections. Early activities in Austria, carried out in the late 1970s, are not included here because Austria has never started up its nuclear power program. Very recent activities in some eastern European countries indicate likely implementation of dry spent fuel storage, but information is currently too sketchy to report here.

11.1 DRY STORAGE ACTIVITIES IN ARGENTINA

In Argentina, the plans have been to store the spent fuel in the reactor pools for a few years, followed by reprocessing in a domestic reprocessing plant. Delays in completing its reprocessing plant (pilot-scale, capacity 5 MTU/yr) have forced the consideration of additional storage capacity, which includes the use of dry storage, and longer time periods (Schneider et al. 1991; Campa 1991).

Further storage may be needed after several years of wet storage for the Argentinean spent fuel from its only reactor types, the 640-MWe CANDU reactor (which started up in 1984) and the similar German-made 357 MWe PHWR (which started up in 1974). Argentina plans to provide storage beyond the reactor pools by dry storage, using the Canadian concrete cask concepts used at the Canadian Gentilly-1 and Douglass Point reactors, but with Argentinean design. The spent fuel will be placed in a number of sealed stainless steel canisters that are then placed in concrete casks. The loaded concrete casks will be placed on a concrete pad at the reactor storage site. The need for additional spent fuel storage at the CANDU reactor requires the initial dry storage activities to start in 1993. A dry storage system at the CANDU nuclear power station is planned. (See Section 4, "Dry Storage Experience in Canada," for spent fuel and storage concept descriptions.) The concept of a central interim dry storage facility is under study (Araoz et al. 1990; NEI 1991).

A dry storage facility with 36 concrete casks is planned for the first five years after starting up the dry storage facility at the CANDU nuclear power station, with the possibility of expanding the capacity to a total of 239 concrete casks. Each cask can store 540 CANDU spent fuel elements (10.2 MTU) in nine steel baskets with a minimum age of seven years since reactor discharge (Campa 1991).

A proposed dry-well storage facility for spent fuel had been under development in Argentina, but the status of this activity is unknown; since receiving the 1990 information on plans for concrete storage casks, it is believed that this activity has been discontinued. The dry-well facility would contain four modules containing 108 storage holes. Each module was to be built into a pit that is 8 m deep, 12 m wide, and 15 m long. The pits are lined with steel tubes, 0.450 m in diameter by 6.1 m deep, with their tops 0.64 m below the concrete surface. Each storage hole would contain seven sealed canisters (arranged with a center canister and six in a surrounding circular pattern) containing the spent fuel assemblies. Each well would be covered by a shielding plug. Cooling was to be provided by forced, circulating air supplied to each well liner through a plenum at the bottom of the pit. The air would flow upward and around the canisters, then exhaust through ventilation holes in the shielding plug covers. The cooling system has been simulated with electrical heater tests (IAEA 1988).

11.2 DRY STORAGE ACTIVITIES IN THE FORMER GERMAN DEMOCRATIC REPUBLIC

The former German Democratic Republic (GDR) was united with the Federal Republic of Germany (FRG) in October 1990. The dry spent fuel storage experience in the former FRG is described in Section 5 of this report. There were six operating nuclear reactors in the former GDR, with a total generating capacity of 2,275 MWe. However, these reactors have all been shut down since reunification with the FRG due to safety and licensing issues. The GDR reactors were all Soviet-designed PWRs (VVERs) (NEI 1991).

Onsite storage of spent fuel at nuclear reactors in the former GDR was conducted in pools, with 250 MTU total capacity. An additional 560 MT of offsite interim storage capacity is available in a central pool at the Greifswald site, commissioned in 1985 and shut down in 1990. The former GDR had sent some spent fuel to the former USSR for reprocessing up to 1985, and had been negotiating with the former USSR on reprocessing the former GDR spent fuel. More recently, negotiations have been under way with France and the U.K. for reprocessing services, and plans are under consideration for a dry storage facility similar to that in the FRG at Gorleben (i.e., a building to house Castor casks for up to 300 to 700 MTU) (Peehs and Rudolf 1991; NN 1991).

The former GDR participated in the IAEA BEFAST-I and -II studies on spent fuel storage. The GDR has performed calculations and experiments with model casks for developing and verifying a computer code on the thermal behavior of spent fuel stored in dry casks. Storage in casks in both the vertical and horizontal positions was studied. Heat transfer processes were studied to allow for prediction of maximum fuel rod surface temperatures. A heated model of a VVER fuel assembly was used in a gas-filled model cask with various cover gases (air, helium, argon). Helium was found to provide the best heat removal. By increasing the cask internal pressure (in the range 0.1 to 0.5 MPa, or 1 to 5 atm) the heat transfer was improved, and the maximum fuel rod temperature was considerably reduced. Under atmospheric pressure conditions, significant differences were not found for the maximum surface temperature of fuel elements in the vertical and horizontal positions. For pressurized conditions, considerable reductions in temperatures were found for the vertical storage position with either air or argon as the cover gas.

These experiments were done for applicability of dual-purpose (i.e., transport and storage) casks (IAEA 1987; Nitsche and Rudolf 1989; Nitsche and Rudolf 1991).

11.3 DRY STORAGE ACTIVITIES IN ITALY

In July 1990, it was decided that the four nuclear power plants (two BWRs, one PWR, and one Magnox) in Italy would remain shut down indefinitely. Italy has decided that most of its LWR fuel (about 243 MTU) will be reprocessed in other countries; disposition of LWR fuel that is considered to be not worth reprocessing (about 153 MTU) is yet to be determined. A design had been produced for a dry cask facility for extended storage of the PWR fuel, but activity has been stopped since the decision was made to use foreign reprocessing. Some specialty fuels (i.e., 64 Elk River fuel elements containing 1.69 MTU-Th and 120 pins from CIRENE and others containing 0.092 MTU-Th-Pu) have been in dry storage, but little more is known. The CIRENE fuel has been stored in a dry condition in a hot cell (IAEA 1990; Guidotti 1991a).

Fuel from the Italian Magnox reactor (Latina) has recently been shipped for reprocessing in the U.K. It had been stored for an unknown period in water, then stored in a dry condition in sealed canisters in a storage pool for a "few years" with an air atmosphere, followed by 15 years with a nitrogen atmosphere. Apparently all the fuel elements had some bare uranium exposed from cladding corrosion [12 to 17% of the total surface area, if the writeup is interpretated correctly], but it was reported that 90% of the fuel elements were in good condition, 8% showed moderate corrosion, and 2% showed severe corrosion (Guidotti 1991b).

11.4 DRY STORAGE ACTIVITIES IN SOUTH KOREA

The Republic of Korea has generated electricity using nuclear power since startup of its first PWR in 1978. The Korean government is pursuing an aggressive nuclear power program to reduce its dependence on foreign fuels. South Korea has one CANDU reactor and eight PWRs (with capacities ranging from 600 to 1000 MWe) in operation, with a total gross capacity of 7.7 GWe. Two additional PWRs are under construction, and one CANDU and two more PWRs are planned (NEI 1991). A total of 14.7 GWe is planned to be operational by the year 2000 and 5,020 MTU of spent fuel will be in storage at that time (Park et al. 1991a).

The final decision on the disposition of spent fuel (i.e., reprocessing and recycle of fuel materials or direct disposal of spent fuel) is not expected to be made in the near future. Thus, additional interim storage facilities away-from-reactors are needed by 1997 after reracking of existing reactor pools is completed and the reracked pools are full. The decision was made for the first extended interim storage facility (away-from-reactors) for PWR spent fuel to be a conventional storage pool, but dry storage is planned at the CANDU reactor station. In the meantime, some transshipment of spent fuel for storage at other onsite nuclear power reactors has started (Park 1990).

For interim storage of the CANDU fuels, the Korean Electric Power Corporation (KEPCO) plans to install 72 concrete canisters for dry storage of approximately 37,800 CANDU fuel bundles (about 700 MTU). To accomplish this, KEPCO has signed an agreement for Atomic Energy of Canada, Ltd. (AECL) to provide the dry storage technology for spent fuel at the CANDU Wolsung-1 reactor. Each dry storage canister can hold nine stainless steel containers, each holding 60 bundles of CANDU fuel, for a total amount of about 10 MTU per canister. The design temperature limit for the design period of 50 years is 180°C. The project to allow for dry storage was expected to be completed in February 1992 (Lim 1991). In the meantime, Korea is undertaking generic R&D related to dry spent fuel storage, including dry rod consolidation (Park 1990; Frost and Wasywich 1991).

A concept for dry storage of spent fuel in a vault-type storage research facility, the Spent-Fuel Technology Advanced Research (STAR) facility, has been investigated in Korea relative to heat transfer considerations. The concept is being considered as an interim storage concept while the spent fuel is awaiting activities such as disassembling, rod consolidation, and special packaging for storage and disposal. The concept consists of placing one PWR fuel assembly in a sealed canister that is lowered into a well (one of 48) below the floor of a hot cell. Hot cell air flows by natural convection from the bottom of each well to the top and exhausts to a chimney (Park et al. 1991b; Lee et al. 1991).

11.5 DRY STORAGE ACTIVITIES IN SPAIN

Spain has nine operable nuclear power plants: seven PWRs and two BWRs. The total Spanish nuclear generating capacity is 7,368 MWe gross. Eleven nuclear power plants previously planned have now been cancelled or indefinitely deferred (NEI 1991).

Spanish reactors have an average of 10 years onsite spent fuel storage capacity in pools with compact racks. Some Spanish reactors expect to fill their present storage pools to capacity as early as 1993 (Ruiz-Lopez et al. 1988; NUKEM 1991). Spain was planning for a central away-from-reactor (AFR) storage facility (combining both wet and dry storage technologies) for 6000 MTU of spent fuel, but the plan has met with some public opposition. The plan called for installation in two stages of 3,000 MTU each, with a first-phase combination of 2,500 MTU capacity in modular pools and 500 MTU in dry metal casks. Spain now plans to use dual-purpose dry storage casks at the reactor sites, then transport them to the final disposal repository site for further dry storage until the repository has been constructed (expected about 2020). Spain's present reference disposal option is for the direct disposal of spent fuel in a deep geologic repository (Schneider et al. 1991; NAC 1991).

Spain previously conducted indigenous development work, starting in 1981, on a dry transport and storage cask for spent LWR fuel assemblies. The project, called "Centauro," was being implemented by the National Uranium Company (ENUSA), the Nuclear Energy Board (JEN), and the Nuclear Equipment Company (ENSA). The Centauro cask was developed to half-scale. The Spanish cask design was to contain 17 PWR fuel assemblies (17 x 17 type) containing 7.8 MTU, with a total maximum cask weight of about 120 MT and maximum dimensions of 5 m long and 2 m in diameter. A maximum fuel cladding temperature was conservatively selected as 250°C. The initial design fuel enrichment was 3.7% ^{235}U, and the design burnup was 40,000 MWd/MT. The fuel was to be cooled for a minimum of five years prior to dry storage. The thermal capacity was 17 kW. Nodular cast iron was initially selected as the cask material, and aluminum was selected for the fuel basket (Melches et al. 1982; Harmon and Johnson 1984; IAEA 1988; Schneider et al. 1991).

Spain recently selected the Nuclear Assurance Corporation's (NAC, a U.S. company based in Atlanta, Georgia) dual-purpose cask (NAC-STC) for its spent fuel storage system. The Electric Power Research Institute (EPRI) in the U.S. is also participating in the program. A Spanish fabricator (ENSA) will construct the NAC-designed casks, which are presently undergoing licensing review by the U.S. Nuclear Regulatory Commission (NRC) and are expected to be licensed in 1992. After the cask is licensed by the NRC, it is expected that the Spanish authorities will also license it. The first cask is expected to be delivered in 1992, a number of casks are needed by 1995, and a central AFR is expected to be built by the year 2000 (Gonzales 1992). The NAC's current design is suitable for fuel from the Spanish Trillo PWR; NAC is in the process of designing a cask for the Almaraz PWRs. The current design has capacity for 26 intact PWR fuel assemblies, and the capacity may be expanded to 56 consolidated assemblies, but lifting weight (about 115 MT) may be a limiting problem. A cask design for BWR fuel is in the planning stages[a] (Estratos 1991; Stuart and Lessor 1992).

Spain has performed research and development on heat transfer for transport and storage casks. Heat transfer in a simulated 4 x 4 fuel rod bundle was studied in both the horizontal and vertical orientations (Palancar et al. 1990). Convective and radiative heat transfer in the cask fins was also studied (Solis et al. 1990).

(a) Nuclear Assurance Corporation (NAC). 1991. Nuclear Assurance Corporation, personal communication with S. J. Mitchell, PNL. March 1991.

11.6 DRY STORAGE ACTIVITIES IN THE FORMER USSR

Nuclear power has been regarded as the best way to provide increased electrical energy needs in the former USSR for a long time. Nuclear power has been aggressively pursued by the government since commissioning of its first demonstration power reactor in 1954. As of April 1991, the former USSR had a total of 47 nuclear power reactors operating with a gross capacity of about 35 GWe. Twenty-eight of these are VVERs (the Soviet designation for PWRs; capacities range from about 400 MWe to the current standard of 1000 MWe), one is a BWR (62 MWe), 16 are RBMKs (the Soviet designation for their water-cooled, graphite-moderated reactors; capacities range from five to the current standard of 1000 MWe), four are FBRs (capacities range from 12 to 600 MWe), and one is an organic-moderated reactor (5 MWe). As of August 1991, the former USSR still planned for the world's largest near-term increase in nuclear power reactors, with 15 under construction and 8 more planned. Strong public concerns in the past few years, however, about the safety of the USSR reactors and about waste management practices have severely reduced the plans for near-term expansion of nuclear power. As a result, a large number of operating reactors have been shut down for safety reasons and others under construction or planned have been cancelled or indefinitely deferred (Kondratyev et al. 1991; Schneider et al. 1991; NEI 1991).

The former USSR stores all of its spent fuel in at-reactor water pools, followed by subsequent storage in centralized pools at multiple-reactor stations where necessary. Spent fuel from VVERs was to be reprocessed, but delays have occurred in the last five years. Also, reprocessing in Russia is now more costly than it was, and many of the former Soviet republics may not have their spent fuel reprocessed (in Russia) as was done previously. The decision on reprocessing of RBMK fuel has not been made; it is stored in water pools at the current time (Spichev and Romanovski 1989; Schneider et al. 1991). The former USSR has made no known commitment for dry storage; however, it is considering extended dry storage, particularly for its RBMK fuel, in "transportation casks." The former USSR purchased two Castor-V transportation/storage casks (one in 1984, one in 1989) for testing (Kritzky 1991; Schneider et al. 1991).

The VVER-440 fuel assemblies consist of about 126 rods arranged in a hexagonal cross-section and containing about 0.12 MTU of UO_2 enriched to about 3.6% ^{235}U. The rods are 9.1 mm in diameter and about 3 m long with a wall thickness of 0.65 mm, and are made of zirconium-1% niobium alloy. Typical burnup is about 29,000 MWd/MTU. The VVER-1000 fuel assemblies (in the more recent Soviet PWRs) consist of about 331 rods arranged in a hexagonal cross-section and containing about 0.437 MTU of UO_2 enriched to 3.3 to 4.4% ^{235}U. The rods are 9.1 mm in diameter and about 4 m long with a wall thickness of 0.65 mm, and are made of zirconium. Burnup ranges from 27,000 to 40,000 MWd/MTU. The spent fuel assemblies from RBMK-1000 reactors (many of these have been shut down and no new units are planned) consist of 36 rods (arranged in an array of 18 rods in a circular pattern, with two sub-assemblies end-to-end within a channel) containing about 0.113 MTU of UO_2 enriched to 1.1 to 2.4% ^{235}U. The rods are 13.5 mm in diameter and about 9 m long with a wall thickness of 0.9 mm, and are made of zirconium or zirconium-1% niobium alloy. Burnup ranges from 18,000 to about 22,000 MWd/MTU (Rybalchenko et al. 1985; NEI 1991).

The former USSR has carried out some dry storage demonstration studies using casks. In one demonstration, VVER-1000 spent fuel assemblies were placed into a Soviet TK-13 transportation cask (considered also to be suitable for short-term storage) after three years of wet storage. The TK-13 cask is made of stainless steel and has a single lid that uses rubber gaskets. Neutron shielding is provided with antifreeze in an outer jacket. The cask is 2.43 m in overall diameter and 6.035 m in overall length, with a cavity of 1.32 m in diameter by 4.95 m long. The cask weighs 116 MT and can hold 12 VVER-1000 fuel elements (about 6 MTU) (Dubrovskij et al. 1987; Kondratiyev et al. 1991). Fuel from the South Ukrainian power plant placed in the TK-13 cask originally had 3% ^{235}U content, it was irradiated to 27,300 MWd/MTU, and had been stored in a pool for 800 days. The heat content of the 12 assemblies in the TK-13 cask was 22 kW at the time of insertion in the cask with an inert cover gas and the cask was stored in the horizontal position. As of the spring of 1991, spent fuel had been stored in the cask for 390 days; the test results indicated the TK-13 cask could be used for short-term storage, and the elastomeric lid seal used would maintain tightness for one year (Kadarmetov 1991).

In the other dry cask storage test, fuel from the Novovoronezh-V power plant was placed in a Castor-V cask with an inert cover gas and the cask was stored in the horizontal position [no other information was given] (Kadarmetov 1991).

In another demonstration, RBMK fuel elements were placed in air storage after 8 to 10 years of in-pool storage. The air storage "caused no serious impact on corrosion resistance and mechanical properties of fuel cladding." From the Soviet studies that include modeling of failure mechanisms, they concluded that the recommended long-term temperature limit be 125°C in dry air and 350°C in an inert atmosphere, as limited by tensile hoop stresses. For RBMK fuel after five-year's storage, the maximum temperature in the center of a dry cask is estimated to not exceed 270°C. From tests and modeling of stress corrosion cracking caused by internal fission products, rods with cracks 80 to 100 μm deep can crack through the cladding in as short a time as two years at 350°C. The "technology of long-term dry storage in transportation casks is under active study now" (Kritzky 1991; Kadarmetov 1991).

A concept was drawn up for regional storage of large quantities of RBMK fuels in sealed stainless cans in channels in a vault cooled by natural convention. Each of the 468 channels in each of the 14 vault sections would hold three cans of fuel stacked end to end. Another concept being proposed is concrete cask storage of RBMK fuels. Each cask would store three cans (as above, 630 mm diameter by 4.0 m high) side by side. Other dry storage systems using concrete casks are under consideration (Kondratiyev 1991; NEW 1991).

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