IRRADIATED FUEL STORAGE: OPERATING EXPERIENCE AND DEVELOPMENT PROGRAMS

INTERNATIONAL WORKSHOP
OCTOBER 17 & 18, 1984

PROCEEDINGS

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Ontario Hydro with U.S. liaison through EPRI

Park Plaza Hotel, Toronto, Canada
ONTARIO HYDRO

PROCEEDINGS OF INTERNATIONAL WORKSHOP ON IRRADIATED FUEL STORAGE: OPERATING EXPERIENCE AND DEVELOPMENT PROGRAMS

October 17 and 18, 1984
Park Plaza Hotel
Toronto, Ontario, Canada

EDITORS: S.J. NAQVI and C.R. FROST

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EDITORIAL NOTE

The papers for these proceedings were typed according to a standard format supplied to the authors by Ontario Hydro. In general, the papers are presented exactly as submitted by the authors, except for minor typographical changes and a few modifications made in consultation with the authors. However in several cases, papers were retyped in part or whole to give a consistent format.

The Proceedings contain all papers presented at the Workshop plus some papers that, although submitted too late, were of sufficient relevance to the Workshop theme for inclusion. Also included is a synopsis of panel discussions, based on written submissions from the panelists.

Summaries of all papers presented were handed out prior to the Workshop to all participants.

Ontario Hydro
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Toronto, Ontario, Canada
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ONTARIO HYDRO

INTERNATIONAL WORKSHOP ON IRRADIATED FUEL STORAGE: OPERATING EXPERIENCE AND DEVELOPMENT PROGRAMS

October 17 and 18, 1984
Park Plaza Hotel
Toronto, Ontario, Canada

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# TABLE OF CONTENTS

Welcome and Introduction (T J. Carter, Ontario Hydro) .................. 1

**SESSION 1 - IRRADIATED FUEL MANAGEMENT STRATEGIES**

Panel Discussion: Irradiated Fuel Management Strategies ............ 5

Spent Fuel Management Strategies with United States
(C.W. Conner, U.S. DOE) ........................................... 24

The French Utility's View on Irradiated Fuel Management
(P. LeSueur, EdF) .................................................. 40

**SESSION 2 - WATER POOL STORAGE**

Operational Experience with Ontario Hydro's Irradiated
Fuel Bays (C.R. Frost, Ontario Hydro) ................................ 51

Water Pool Operational Experience at Morris
(E.E. Voiland, et al, GE) ............................................. 80

Storage of Irradiated AGR Fuel in CEGB Reactor Ponds
(K.A. Simpson, CEGB) ............................................... 100

Current Design of Waterpool Type ISFSI
(W.L. Dobson, Gilbert-Commonwealth) ................................ 108

Storage of Irradiated Fuel in British Nuclear Fuels
plc (BNFL) Ponds (D. Elliott, BNFL plc) ................................ 116

**SESSION 3 - DRY STORAGE TECHNOLOGY AND ENGINEERING STUDIES**

Dry Storage - Past, Present and Future
(A.B. Johnson, Jr. et al, PNL) ....................................... 120

Onsite Storage of Spent Nuclear Fuel in Metallic Spent
Fuel Storage Casks (M.L. Smith, et al, Vepco) ...................... 145

Decay Heat and Heat Transfer Predictions for Spent Fuel
Storage Systems (J.M. Creer, PNL) ................................... 157

Canadian Experience with Concrete-Canister Dry Fuel Storage
(R.O. Sochaski, AECL) ............................................... 211

Engineering and Safety Features of Modular Vault Dry
Storage (D. Deacon, et al, GEC-ESL) .................................. 232

Dry Storage Systems Evaluation at Ontario Hydro (J. Freire-Canosa, et al, Ontario Hydro) ................................. 277

Present Status of the Development of a Dry Spent Fuel Storage Cask in Spain (A. Uriarte, et al, JEN) ...................... 301

Dry Storage Facility for Spent Fuels in Japan (M. Adachi, et al, JAERI) ............................................................... 311

SESSION 4 - DRY STORAGE ECONOMICS, STANDARDS AND LICENSING


Status of United States Standards for Spent Fuel Storage Activities (J. Nevshemal, Westec) ................................ 348

AGR Dry Fuel Storage - Design for Acceptance (L.J. Saunders, CEGB) ................................................................. 355

Licensed At-Reactor Horizontal Concrete Silo Dry Storage Demonstration Program (R.K. Kunita, et al, CPL) .............. 368

Discussion Period ........................................................................ 390

SESSION 5 - DRY STORAGE - FUEL BEHAVIOUR (I)

Long-term Behaviour of Irradiated CANDU Fuel in Concrete Canister Storage - Test Results (K.M. Wasywich, et al, AECL) ............. 393

Fuel Rod and Crud Behaviour under Long-term Dry Fuel Storage Conditions (C.S. Olsen, EG&G Idaho) .................... 432

An Overview of Dry Interim Storage Research and Development Activities at the E-MAD Facility Nevada Test Site (J.B. Wright, et al, WEC) ......................................................... 456

Iodine-and Cesium/Cadmium-Induced Cracking of Irradiated Zircaloy Cladding at 373 – 573°K (J.C. Wood, et al, AECL) ...... 468

A Probabilistic Prediction of the Allowable Dry Storage Cladding Temperature (T.Y. Cheung, et al, Stanford University) 493

SESSION 6 - DRY STORAGE: FUEL BEHAVIOUR (II)

UO₂ Fuel Oxidation in Air Below 350°C (P. Wood, CEGB) ................................................................. 537
Oxidation of UO₂ at 150°C to 350°C  
(E.R. Gilbert, et al, PNL) ........................................ 551

Effect of Cladding Defect Size on the Oxidation of Irradiated  
Spent LWR Fuel Below 360°C (R.E. Einziger, et al, WHC) ........... 599

Behaviour in Ai. at 175°C to 400°C of Irradiated  
UO₂ Fuel (I.J. Hastings, et al, AECL) .............................. 626

SESSION 7 - DRY STORAGE: THE FUTURE (PANEL DISCUSSION) ........ 655

AUTHOR INDEX .............................................................. 666

WORKSHOP PARTICIPANTS .............................................. 667
Good Morning Ladies and Gentlemen. On behalf of myself, Tom Carter, and Ontario Hydro, I would like to welcome you to Toronto and to this workshop which will take place over the next two days. I would like to say something about Ontario Hydro for those who are not familiar with our company. We are a publicly-owned utility serving the approximately eight million people of Ontario. We have fossil, hydraulic and nuclear generation capability. Currently we have 7,360 MW(e) of CANDU nuclear capacity in-service and are working towards a total of 14,300 MW(e) by 1992. Ontario Hydro carries out its own design and construction as well as operation of its nuclear stations and associated nuclear facilities such as heavy water plants, radioactive waste storage facilities and radioactive materials transportation systems.

The subject of this workshop, 'Irradiated Fuel Storage', is not a new subject but it is a subject which in recent years has acquired new significance and greater dimension as a component in the closing of the nuclear fuel cycle. It is clearly a complement to nuclear reactor operation and to any back-end fuel cycle activities such as recycle and disposal. Although more than two decades of successful experience with water pool irradiated fuel storage is in hand in many of our countries, and this leads to the conclusion that it can be used for several more
decades safely and economically, it's important to recognize that better and more economical methods and technologies for storing irradiated fuel continue to be developed.

Many countries are facing the need to store irradiated fuel beyond the time frame which was originally envisioned in the design of the nuclear power stations. This means that the irradiated fuel storage bays associated with the stations are becoming full before disposal or other fuel cycle facilities are available. As a near-term solution, many utilities have implemented reracking of irradiated fuel, the use of higher density storage modules, and also, fuel rod consolidation techniques. All of these require increased handling of the fuel assemblies and the fuel elements. It potentially also means higher temperatures and incremental radiation effects on bay materials, criticality considerations in some cases are important and there are some regulatory considerations to be overcome. This approach to increasing storage capacity is based on the premise that the cheapest capacity that one can provide is better utilization of that which already exists.

Intermediate-term solutions are based on the construction of more at-reactor storage capacity once the maximum use possible has been made of the existing original capacity. Depending upon the design of this new capacity, and the regulatory jurisdiction, this intermediate-term solution may be somewhat more difficult to implement from a regulatory and public review standpoint than the near-term solutions mentioned above. Auxiliary storage bays with underwater or cask interbay transfer systems are now in use; dry storage vaults have been extensively studied and are in use in the United Kingdom. Moveable canisters of various materials have been developed and demonstrated and placed in limited use, particularly in the Federal Republic of Germany. The use of dry storage systems imposes different performance requirements on irradiated fuel and makes necessary assessment programs to evaluate the long-term integrity of fuel in dry, moist or inerted atmospheres. It
makes it necessary to assess fuel performance at higher temperatures than
is the case of water pool storage systems. Also, the extrapolation of
stored fuel behaviour to time periods of 50-75 years is receiving a good
deal of attention in order to bring our level of confidence about
long-term behaviour in a dry environment up to that which we have for wet
storage environments.

Long-term solutions are based on the disposal of irradiated fuel
or, alternatively, recycle of irradiated fuel with disposal of the
immobilized high level waste. All countries have programs underway
either to develop a concept for disposal, develop and demonstrate a
concept, develop and implement two or three disposal systems, or, indeed,
some combination of the above. Some programs are on very firm schedules
right to the repository in-service date. Some programs are less fixed
with implementation dates not yet determined or they are far in the
future. One question that many countries face is that of the resource
value of irradiated fuel and when, if at all, to implement recycle prior
to disposal. These aspects, particularly the acquisition of regulatory
and social approvals and the recycle question, make it prudent to plan
for the eventuality of extended storage of irradiated fuel, perhaps
always at reactor sites or in some centrally located regional or national
facilities. Very long term, say 100 years, durability of fuel in storage
environments and durability under transportation conditions at some far
future date are important considerations. Experience and research
programs to date indicate that pool storage of irradiated fuel for long
time horizons is practical both from the fuel integrity and pool design
standpoints; we will hear about this in Session Two this morning.
Research into dry storage behaviour has not been carried on so long, but
substantial knowledge has been gained over the last several years in a
theoretical and practical sense. Dry storage and its practical
implementation as part of a fuel management strategy is a central theme
of this workshop. This is one of the more, possibly, optimum solutions
that I referred to previously.
Ontario Hydro is approaching a juncture, a decision about which kind of additional storage capacity to build next, to meet our needs in the 1990's. We have been carrying out conceptual engineering and research programs in anticipation of the pending decision, and we felt this would be an excellent time to meet with other international experts in a workshop forum where all of us can discuss our programs, results and future expectations. I will be particularly interested in hearing viewpoints about the use of a multi-purpose or "once-through" container that can serve as a storage, transportation and disposal package. It offers fuel handling simplicity, but at what cost and under what practical limitations? Does the multipurpose role impose extra demands on fuel element performance not required in a storage-only, dry mode? What role does this integrated package concept have as an optimum long-term irradiated fuel storage option?

On a more social note, I would like to remind you of the reception this evening and the banquet. For those of you visiting Toronto for the first time or for the first time in a long while, I hope you will have time to explore a few of the many attractions our city has to offer. Those of us who live in Toronto are certainly proud of it. You'll find two brochures in your registration kits that might prove helpful during your visit. This year is the 150th birthday of Toronto and there are celebrations throughout 1984. The gentleman who is in charge of the birthday celebrations for Toronto will be speaking at the banquet tonight. He is, among other things, a well known Toronto historian and an excellent speaker. He will certainly identify some major features of Toronto that you might be interested in visiting.

At this point, I would ask Mr. Bob Williams of the Electric Power Research Institute to open Session One. One again, welcome to this workshop.
SESSION 1

IRRADIATED FUEL MANAGEMENT STRATEGIES

Chairman:  R.F. Williams
             Electric Power Research Institute
Spent fuel storage is the common element in all fuel management strategies.

1. Regardless of a country's overall fuel disposition strategy, storage is a primary element.

2. Delays in reprocessing or disposal options will almost certainly increase the demand for storage.

3. Storage is the near term need requiring decisions now.

National policy is a major variable.

1. Policy toward reprocessing is major factor.

2. Government's role in overall fuel disposal strategy has become a dominant factor.
   (a) U.S. NWPA is key example.

3. Responsibility to meet storage needs has remained principally with utilities.
   (a) Technology Choice;
   (b) Licensing;
   (c) Implementation Schedule.

4. In all cases the total cost of fuel management will be paid by the users.

Utilities are faced with a series of dilemmas.

1. How much do they dare rely on planned government programs?

2. What is the proper "prudent management" planning base?

3. Will near term storage actions put utilities at a disadvantage in the future?
   (a) Impact of rod consolidation;
   (b) Could fuel lose place in removal queue?
   (c) Could utility incur unnecessary near term costs?

Utility decision is simplified by striking the correct balance between minimizing costs and having assurance of adequate storage.

1. How much insurance do utilities need and can they afford it?
2. Given the uncertainties of governmental actions, can utilities plan with any degree of confidence?

3. Will public perceptions become a deciding factor in deciding utility strategies?

Range of storage alternatives appears to be available:

(a) Pool reracking and new pools;
(b) Casks;
(c) Silos;
(d) Vaults;
(e) Dry Wells.

Demonstrations are still needed however to establish:

(a) Capital cost economics;
(b) Operational characteristics and costs;
(c) A complete data base for generic licensing;
(d) Licensing procedures.

There also remain technical issues in the areas of safety margins, operational procedures, and storage economics, as follows:

(a) Fuel integrity over the long term;
(b) The need for and methods of monitoring advanced storage technologies;
(c) A sensible approach for handling and storing failed fuel;
(d) A better understanding to the role played by storage canisters;
(e) Improved techniques for the treatment and disposal of bundle scrap hardware in fuel consolidation;
(f) Development of storage systems that can be favorably integrated with longer term technology choices.
SESSION 1 PANEL DISCUSSION: IRRADIATED FUEL MANAGEMENT STRATEGIES

Panelists

H.N. Isaac          Ontario Hydro
C.W. Conner         United States Department of Energy
D. Deacon           GEC Energy Systems Limited (United Kingdom)
J.B. Moegling       Tennessee Valley Authority (United States)
P. Le Sueur          Electricite de France
B. Gustafsson       Swedish Nuclear Fuel Supply Co.

EDITORS NOTE: Time allotted to this Panel Discussion, plus follow-up Question Period was 75 minutes.
In this ten minute contribution to the panel discussion I will present an overview of the options being considered by Ontario Hydro for used fuel management.

As a first step in our storage program we are maximizing the used fuel storage capacity in our existing water pools at the reactor sites. For Pickering, this is being achieved by introducing this year and through 1985 a new standard storage container called a module (Figure 1 shows the various used fuel containers in use in our water pools) which will provide about 50 percent more fuel storage per unit volume of storage space available. For Bruce, new tray stacking frames will provide closer and higher stacks of used fuel storage trays.

What about the future? Figure 2 shows the options that are being considered for the future storage and ultimate disposal of used fuel. Down the left-hand side are approximate dates for these activities. In 1987 we must select the additional storage system for in-service 1994/95 at the Pickering and Bruce nuclear stations.

Under Option 1 (Reference Option), the additional storage will be more water pools at each station.

For Option 2A we are considering dry storage convection vaults or concrete canisters as storage systems only.

For Option 2B we are again considering dry storage in what we term dual purpose metal or concrete casks. These casks will serve as storage containers, also be transportable and possibly be used as the disposal container as well. Our intent is to minimize handling of the fuel.

I have also shown fuel recycle and disposal on this overhead since these activities will form an integral part of our long-range used fuel management program. The dates shown are approximate and represent the earliest dates that these activities can take place providing the appropriate approvals are obtained.

My staff and the staff from AECL will be presenting details on these storage systems later in the program.

* No written contribution to the Panel Discussion was provided by C.W. Conner (USDOE) and J.B. Moegling (TVA).
Storage Containers

Pickering GS
Fuel Storage Basket

Shipping and Storage Module

Bruce GS
Irradiated Fuel Storage Bay

FIGURE 1
ONTARIO HYDRO USED FUEL MANAGEMENT PROGRAM

1984/85 - Current storage on-site in water pools - Action taken to maximize storage capacity using higher density storage modules.

FUTURE OPTIONS

1
Water Pools
Fuel Transportation in Metal Casks
1987
(Select Additional Storage System - In-Service 1994/95)

2A
Dry Storage
- Convection Vaults
- Concrete Canisters
Fuel Transportation in Metal Casks

2B
Dry Storage in Transportable Dual Purpose Casks
- Metal
- Concrete
Fuel Transportsed in Dual Purpose Casks

2010
(Earliest Date For Disposal)
Disposal of Fuel (Recycle Not Considered)

2030-2050
(Earliest Date For Recycle)
Fuel Recycle
Disposal of Waste

FIGURE 2
The Management of Irradiated Fuel has been approached by various countries in different ways.

Great Britain and France have both pursued a policy of reprocessing since the early days of nuclear power, starting with Magnox fuel and latterly building new facilities to deal with irradiated oxide fuels. Both countries offer these facilities to overseas customers, such as Japan and Italy.

In the USA, reprocessing for the civil reactor programme has not been available for many years and the utilities are currently operating on the basis of "at reactor" storage in pools. I am sure that you are all aware of the current MRS programme in the U.S.A.

Canada has always pursued a policy of long-term storage of its irradiated fuel at the power stations and it does not have a commercial reprocessing capability.

Several European countries are currently proposing "once through" cycles involving storing irradiated oxide fuel unreprocessed in appropriate repositories.

The primary advantages of reprocessing are:

(a) To separate materials that can provide fuel for future reactor systems.

(b) To segregate the long-lived fission products for eventual safe storage, i.e., vitrification.

The primary disadvantage to reprocessing is the extremely high cost of the process which in the UK is currently charged to the plants in operation. It could be argued that this cost should be borne by the fast reactor programme and not superimposed upon the operational cost of current reactors. Published information demonstrates that the addition of these reprocessing costs to the present fuel costs increases very significantly the generation cost with no benefit to the current power programme.

With the full-scale commercial fast reactor programme still some way off and other potential methods of producing electricity on the horizon, it would appear to make sense to reserve the option on whether to reprocess or not by storing the irradiated fuel compactly and safely until it can be more clearly seen whether or not the potential fuel value of the plutonium will be realized by the fast reactor route.
Whatever method of storage is chosen and whether it be for intermediate or long-term, I believe a comment made by Mr. Bob Davidson at a meeting of the ANS/ENS in Washington in 1980 still to apply. He said the nuclear industry "wants innovation like a hole in the head".

When comparing the many storage options that are on the market, there are many basic questions to be addressed.

I have selected six that I believe to be most important:

1. Has it been done before? Have the technological principles and safety features been demonstrated on similar plant and what is the operational experience?

2. Can it be licensed and built in an acceptably short timescale?

3. How much did it cost to build and what are the operational costs?

4. Can the fuel be easily retrieved?

5. Can the condition of the fuel and its containment envelope be easily monitored?

6. Are the fuel temperatures in storage such that an inert gas is necessary for satisfactory storage or is the cooling arrangement such that lower temperatures allow air to be used as the containment environment?

In the UK we have had significant experience of the design, construction and operation of several dry store facilities.
The basic principle of the Electricite de France strategy is irradiated fuel reprocessing. Obviously, it is not an end by itself and it supposes the use of the plutonium, that is to say the development of the fast breeder reactor technology.

Electricite de France has begun to cope with this technology in cooperation with the French Commissariat a l’Energie Atomique with Phenix reactor, for which we have recently celebrated the 10th year of operation. Now, Electricite de France is participating in the design and construction of Super Phenix on the Rhone river, with foreign partners. For this reactor, Electricite de France supplied approximately 3 tons of plutonium for the first core and will supply 1.2 tons for each reload.

As a reciprocity, Electricite de France will participate in the design and construction of the large commercial German fast breeder reactor SNR 2. For the future, and after one or two years of operation of Super Phenix, if possible without major incident, we hope that the French government will authorize Electricite de France to build another fast breeder unit with a power level of about 1,500 MWe. So, we have the potential to use a lot of plutonium. If this use in fast breeder reactors becomes decreasing, we would be ready to recycle plutonium in the PWR units without changing our strategy.
1.0 THE SWEDISH NUCLEAR POWER PROGRAMME

1.1 1983
(a) 10 units in operation corresponding to 7500 MW.
(b) Nuclear electricity contribution ~37%.

1.2 1986
(a) 12 units in operation corresponding to 9500 MW.
(b) Nuclear electricity contribution 45-50%.

1.3 2010
(a) According to the parliament resolution none of the reactors shall be operated beyond the year 2010.

2.0 QUANTITIES OF RADIOACTIVE WASTE WITH FULL UTILIZATION OF THE TWELVE-REACTOR PROGRAMME

(a) 7,500 metric tons (calculated as uranium) of spent fuel.
(b) 100,000 m³ of low- and medium-level waste from reactor operation.
(c) 150,000 m³ from decommissioning waste.

3.0 SPENT FUEL STRATEGY

(a) Around 870 tonnes are presently covered by foreign reprocessing contracts.
   140 tonnes with BNFL.
   730 tonnes with Cogema.

(b) Remaining 6,630 tonnes actually foreseen to be disposed of direct after 30-40 years intermediate storage in the CLAB AFR facility.
THE SWEDISH NUCLEAR SYSTEM

URANIUM ORE

DOMESTIC URANIUM MINING

NATURAL URANIUM 32,000 tons U

ENRICHMENT 7,000 tons U

STOCKPILE ENRICHED URANIUM

FUEL ELEMENT

12 NUCLEAR POWER STATIONS

1500 TWh EL. ENERGY

SPENT FUEL 7,000 tons U

WASTES FROM OPERATION 100,000 m³

WASTES FROM DECOMMISSIONING 150,000 m³

SEA TRANSPORT SYSTEM 1982

850 tons URANIUM 6 tons PLUTONIUM

CLAB 1985

REPROCESSING CONTRACTS 867 tons

INTERMEDIATE WASTE STORAGE 1990

ENCAPSULATION AND TERMINAL STORAGE, SFL 2020

REPOSITORY SFR REACTOR 1988
4.0 SYSTEM IN OPERATION AND UNDER IMPLEMENTATION

4.1 Sea Transportation System

(a) Special ship commissioned in 1982 and used for transportation of spent fuel to the reprocessing plant at la Hague, France.

(b) Ten transport casks of the type TN17 Mk2 (French design). All of them delivered and partly used.

(c) Two transport casks of the type TN17 Mk2 to be used for transport of core components.

(d) Three specially designed vessels. Two are in use and the third under manufacturing.

(e) Total investment 200 MSEK (1 US$ = 8.5 SEK).

4.2 A Central Facility for Interim Storage of Spent Fuel

(a) Storage Capacity

3,000 tonnes of fuel (as uranium). Can be extended up to 9,000 tonnes.

(b) Receiving Capacity

300 tonnes/year.

(c) Concept

Wet unloading and auxiliary systems in surface located facilities. Wet storage in an underground rock cavern.

(d) Number of Receiving Lines

2 (~100 casks/year).

(e) Number of Storage Pools

4 + 1 in reserve.

(f) Cooling Capacity

6.5 MW.

(g) Cooling Water Flow

300 litres/sec.
(h) **Special Design Features**

Storage in rock cavern, pools withstand boiling and earthquake (0.1 g), remote maintenance etc.

(i) **Start of Construction**

1980.05.01.

(j) **In Operation**

Middle 1985.

(k) **Total Construction Cost in Current Money**

1,700 MSEK.

(l) **Operation Cost**

~65 MSEK/year.

4.3 **A Central Final Repository for Low- and Medium-Level Reactor Waste - SFR**

(a) **Storage Capacity**

50,000 m³ which cover the need up to year 2000. Extension planned up to 110,000 m³.

(b) **Concept**

In crystalline rock under the Baltic Sea adjacent to the Forsmark nuclear power station.

(c) **Start of Construction**

1983.07.01.

(d) **In Operation**


(e) **Total Construction Cost**

950 MSEK.

5.0 **FUTURE SYSTEMS**

Extensive R&D on the system of direct disposal of non-reprocessed spent fuel, the so called KBS-3 method. Construction of the final repository is planned to start around the year 2010. The time available until then will be used for further R&D work, optimization and design, site
investigations and site selection. An application for licensing of a repository is foreseen around the year 2000.

6.0 COST AND FINANCING

(a) Financing

Back-end activities are financed through a fee on the nuclear power production. For 1984 the fee is 0.019 SEK per kWh (1 US$ = 8.5 SEK).

(b) Cost

If the reprocessing costs are excluded, the distribution of costs is roughly as follows:

<table>
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<th>Activity</th>
<th>Cost Percentage</th>
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<tr>
<td>Transportation</td>
<td>7%</td>
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<tr>
<td>Construction and Operation of CLAB</td>
<td>20%</td>
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<tr>
<td>Encapsulation and Final Disposal of Spent Fuel</td>
<td>38%</td>
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<tr>
<td>Decommissioning</td>
<td>35%</td>
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7.0 REFERENCES


OFFICE BUILDING
Offices, canteen, washing- and dressing rooms.

AUXILIARIES
Cooling and clean up systems, ventilation, waste handling.

FUEL ARRIVAL

ELECTRICAL BUILDING
Electrical power and control systems.

FUEL ELEVATOR
Transport of spent fuel between reception and storage sections.

FUEL RECEPTION
Equipment for receiving, cooling, cleaning and unloading of transport casks.

GARAGE AND STORES
Garage and maintenance shop for transport vehicles.

FUEL STORAGE
The spent fuel is stored in water filled pools.

CLAB
OSKARSHAMN POWER STATION AND CLAB

1. Oskarshamn 1
2. Oskarshamn 2
3. Oskarshamn 3
4. Gas turbine
5. Cooling water intake O1-O2
6. Cooling water intake O3
7. Restaurant
8. Construction office
9. Workshop
10. Parking
11. Entrance to the rockstore for waste
12. Harbour
13. Exhibition
14. Hotels
15. Cantonment
16. Meteorological tower
17. Switchyard
18. Central storage facility for spent nuclear fuel (CLAB)
19. Landing-stage
20. Main gate to the fenced area
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| INSTALLATION:           |        |        |        |        |        |        |
| FULL RECEPTION          |        |        |        |        |        |        |
| AUXILIARY               |        |        |        |        |        |        |
| ELECTRICAL              |        |        |        |        |        |        |
| FUEL STORAGE            |        |        |        |        |        |        |

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SESSION 1 PRESENTATIONS

SPENT FUEL MANAGEMENT STRATEGIES

IN THE UNITED STATES

Carl W. Conner

U.S. Department of Energy
United States of America

ABSTRACT

With the passage of the Nuclear Waste Policy Act (NWPA), a clear policy was established for the management of spent fuel and high-level wastes in the United States. The NWPA is a result of years of intensive effort on the part of Government, industry and concerned citizens to seek an acceptable solution to the problems associated with the management of nuclear wastes.

The NWPA establishes policy, division of responsibilities and a time schedule for the management of wastes both in the near-term and in the long-term. The policy is that the U.S. Government will take title to the waste, beginning in 1998, and dispose of it in a geologic repository. An alternative to that policy is that the Government could, if authorized and directed by the Congress, store the waste in a Monitored Retrievable Storage (MRS) facility.

The utilities are responsible for near-term storage of the spent fuel (prior to 1998) and the Government is responsible for accepting spent fuel beginning in 1998. The Government is also responsible for the transportation of the waste beginning in 1998. The cost of the program is to be borne by the utilities.

The challenge for the government is to design a system for accepting the waste, that is safe, efficient, cost-effective and can be implemented in a timely manner. The Government is working with the utilities, designers of nuclear facilities, State and local officials, public interest groups, the Congress and private citizens to plan and implement a system that best meets the objectives of the NWPA.
SPENT FUEL MANAGEMENT STRATEGIES
IN THE UNITED STATES

In the United States, there are currently about 80 operating reactors producing about 300 terawatt hours of electricity and discharging about 1,300 MTU of spent fuel this year.

Since the first commercial power reactor became operational in 1957, the management of spent fuel has consisted essentially of storing it in pools at the reactor site. There are about 11,000 MTU of spent fuel currently being stored in this manner. Up to 40,000 MTU of spent fuel will be discharged by the year 1998, and to 124,000 MTU by the year 2020.

It was originally envisioned that spent fuel would be reprocessed to recover valuable uranium and plutonium. In fact, in 1956 the U.S. Atomic Energy Commission guaranteed to buy all plutonium produced by commercial power plants. Despite efforts by industry to build and operate reprocessing plants at Barnwell, South Carolina, Morris, Illinois and West Valley, New York, commercial reprocessing is not anticipated on any wide scale in the U.S. in the near future. The reasons for this situation appears to be both economic and political. The cost of uranium and demand for separated plutonium have not escalated as much as originally estimated, significantly affecting the economics of reprocessing. The political history of reprocessing has fluctuated from Federal Government encouragement of commercial reprocessing to a total ban by President Carter in 1977. President Reagan lifted the ban in 1981 but charged industry with the responsibility of taking the lead on reprocessing. The Government is trying to help by eliminating regulatory impediments while at the same time ensuring that there are no compromises in safeguards.
The lack of reprocessing has created a problem for many U.S. utilities who had planned their spent fuel storage needs based upon reprocessing beginning by the 1970's or 80's. Some reactor spent fuel pools are filling up and several utilities will be facing storage problems as early as 1986 unless remedial actions are taken. Such remedial actions may include reracking to achieve higher density in existing pools, transshipments of spent fuel to other reactor pools, consolidation of spent fuel rods to achieve maximum density in existing pools and the addition of dry storage in metal or concrete casks at the reactor site.

Reracking is a licensed method of increasing storage capacity. Utilities are expected to utilize this method to the extent practical. In fact, many utilities have already begun reracking. Off-site transshipments of fuel to other pool facilities will occur on a limited basis where reracking cannot be used and no institutional impediments exist.

Disassembly of fuel assemblies and consolidation of the fuel rods is an attractive concept for achieving the maximum density of storage in existing pools. Also, dry storage of fuel in metal or concrete casks is of great interest. These two concepts are mentioned in the Nuclear Waste Policy Act and will be discussed in more detail later.

Another complication in the management of spent fuel has been the Federal Government's inability to provide definite plans for either long-term storage or permanent disposal of spent fuel and high-level waste. In the 1970's several concepts were studied, but it was not until the passage of the Nuclear Waste Policy Act of 1982 that the U.S. finally got its "act" together.

The Nuclear Waste Policy Act of 1982 (NWPA) not only established a U.S. nuclear waste management policy and committed the Government to start accepting waste in 1998, it established a framework for a complete
waste management system for the back-end of the fuel cycle. This waste management system consists of several components including a geologic repository, a possible second repository, one or more possible Monitored Retrievable Storage facilities (MRS), near-term at-reactor storage by the utilities, Federal Interim Storage (FIS) for not more than 1900 MTU, spent fuel research and development, transportation of the waste, as necessary.

The individual components of the system are not new concepts in themselves, but the combination of these components into an interrelated system is unique. Research on geologic disposal in various media has been going on for several years. You may recall the salt project at Lyons, Kansas and the WIPP project near Carlsbad, New Mexico. Interim retrievable storage facilities have been studied for several years, also.

At-reactor storage, until recently, has focused upon pool storage. Not much research has been done on dry storage at reactor sites. There have been some shipments of spent fuel but not of the magnitude envisioned by the NWPA. The unique feature of the waste system as identified in the NWPA is the interrelationships between the various components and the various scenarios that are possible. Let's look at each one briefly.

The NWPA clearly places the responsibility for near-term management and storage of spent fuel with the utilities. The Government is only responsible for accepting and storing not more than 1900 MTU, and then only after a utility or utilities request Federal Interim Storage and the NRC determines that the utilities have done all that can reasonably be expected in trying to solve their own near-term storage problems. The cost of Federal Interim Storage, if needed, will be borne by the utilities that use it. The only at-reactor method of spent fuel storage currently licensed by NRC is pool storage. Adding additional pool storage would be expensive and would require a long lead commitment.
in order to have it ready when needed. Utilities that anticipate a need for only a small amount of additional storage will be reluctant to construct a new pool.

Dry storage in casks offers more flexibility in storage capacity without a long-term commitment to a major construction project. Dry storage in casks, however, has not been licensed by NRC. The NWPA provides for Federal Government assistance to utilities in gathering sufficient information such that dry storage can be licensed. DOE recently selected and is negotiating co-operative agreements with Virginia Electric Power Company and Carolina Power and Light Company to demonstrate dry storage methods and obtain sufficient information for the utilities to license this storage concept.

Dry storage in casks offers several potential advantages, not only to the utility but to the whole waste management system. For the utility, dry storage casks can provide modular storage to fit any storage need from a few metric tons to several hundred, and once licensed, can be implemented in a short timeframe.

Dry storage casks may also be shippable. If shippable dry storage casks can be licensed, the utilities can reduce handling costs by only loading the casks once for storage and not have to unload and transfer fuel to a transport cask for transportation off-site at a later date. This advantage can also apply to other possible storage locations in the waste system. I will talk more about this concept later.

Federal Interim Storage will only become a reality if a certified need arises, and even then the maximum amount of spent fuel that will be stored is 1900 MTU. This is such a small amount of fuel that it is not considered to have a major impact on the design and operation of the whole waste management program.
Monitored Retrievable Storage as currently envisioned in the Mission Plan is essentially a backup or an option to the repository and therefore should be capable of storing as much waste as the repository and have the same capabilities for receiving, packaging and handling waste. Storage in the MRS will be one of several dry storage concepts: above-ground metal storage casks, concrete silos, vaults or below-ground drywells or tunnel racks. The first two methods have been selected for further study.

Repository disposal may be in one of several geologic media: salt, basalt, tuff or crystalline rock. There are seven salt sites being studied in either domed or bedded salt. These sites are located in the states of Utah, Texas, Louisiana and Mississippi. One basalt site is being considered in the state of Washington, and one tuff site is being studied in Nevada. No specific crystalline rock sites have been selected for study, however, formations in 17 eastern states have been identified. Each media has its own unique characteristics that will influence the design of the repository, the emplacement methods and the emplacement package. Current concepts call for spent fuel rods to be consolidated and placed in metal canisters with possible overpacks and then placed in vertically or horizontally drilled holes and sealed with a special material, if necessary. The design of the emplacement package will depend upon the geology selected. The ability to retrieve the waste for a period of time is also a requirement.

Another possible component of such a system is a spent fuel reprocessing facility. The NWPA is essentially silent on this subject. As mentioned previously, President Carter banned commercial reprocessing in 1977. President Reagan lifted this ban but specified that the private sector must take the lead. The nuclear industry has not responded in a definitive manner and it is uncertain at this time whether there will be any significant reprocessing of spent fuel in the near future. There are no current plans for commercial reprocessing and, even if a decision is made to reprocess, it will be five to ten years before it can begin.
Therefore, we do not consider any significant amount of commercial high-level waste in the waste management system concepts and are focusing upon spent fuel rods as the primary waste form.

The last component of the waste management system is transportation. Transportation of waste may occur between several locations in the waste management system covering thousands of miles, and may be by truck, rail, or barge. The NWPA specifies that the private sector must be utilized to the maximum extent possible. A business plan is now being developed to define how DOE will work with the transportation cask designers, carriers and affected States to fulfill the intent of the NWPA.

When looked at individually, each of these components of the system has waste management needs and requirements that can be optimized in itself. However, when looked at as components of a total system, the optimum waste management system may not be just the sum of its parts.

Now that we have defined the various possible components of the waste system as outlined by various requirements and responsibilities of the NWPA, let's look at the possible systems scenarios and how they may influence the overall management of spent fuel. Figure 1 indicates the possible systems scenarios.

The NWPA discusses two methods for expanding at-reactor storage capabilities. One method is rod consolidation. This method would essentially allow for an approximate doubling of fuel rods that could be stored in existing pool space. DOE is negotiating a co-operative agreement with Northwest Utilities Service Company for participation in a demonstration of spent fuel rod consolidation techniques at a wet storage basin. Some pools, however, are not structurally capable of storing the additional fuel. The other method of increasing at-reactor storage is the use of dry storage in metal casks as discussed previously above. The NWPA provides for the Government to work with utilities in co-operative
research and demonstrations to obtain sufficient data for the licensing of dry spent fuel storage in metal casks. The fuel can then be stored in these casks until it is shipped off-site to either a repository, an MRS or to some other facility.

I would like to point out the number of packaging and handling steps involved in the various scenarios. In the "reactor pool to dry storage to repository" scenario (Figure 2), fuel would be handled for loading and unloading three times and placed in three types of containers.

In the scenario that includes an MRS (shown in Figure 3), fuel is handled five times and placed in four different types of containers.

We believe that economies can be made in the system as a whole. These economies may include standardization of the spent fuel package which will allow standardization of handling facilities at all locations. It may be possible to go much further and develop shippable storage casks that could be used for dry storage at the reactor site and then shipped to a MRS, stored and then shipped to the repository (see Figure 4). This would reduce the packaging and handling of fuel to the initial packaging at the reactor pool and the repackaging for repository emplacement. Some have even envisioned possible repository emplacement of the shippable storage cask itself, reducing the packaging and handling of fuel to the initial step at the reactor pool.

Reactors have varying capabilities for the handling and shipping of spent fuel. Some reactors may not be able to consolidate in their pools. Others may be limited in crane capacity or rail access. There are various possibilities for overcoming these problems in most cases. Portable dry consolidation facilities could be moved about and provide rod consolidation as a service. Handling and rail capabilities could be upgraded as necessary. Rail facilities may be upgraded or even added. Barge transport options may also be viable.
Another option would be to centrally locate a packaging and handling facility near the reactors. This facility could receive spent fuel in any form from the utilities, thereby reducing the need of specialized packaging at the reactor site. The fuel rods could then be consolidated and packaged as necessary to best fit the needs of the remainder of the waste system. Limited storage capability could be provided at this facility to optimize the management of receipts from utilities and shipments to the repository.

It may be possible to package waste at this facility for direct emplacement in the repository. Transportation from this facility could be managed, possibly utilizing special unit trains or, if navigable waterways are available, unit barge shipments. Such a facility could offer many more options and provide more flexibility to the whole waste management system.

In addition to the technical aspects of the waste system, there are also institutional issues to be considered. The NWPA requires that the utilities pay essentially all costs associated with waste storage and disposal. The Waste Fund has been established and utilities are now paying 1 mill (one-tenth of a cent) per kilowatt hour into this fund. This is in addition to any costs of utility at-reactor storage or Federal Interim Storage that may be necessary for the near-term.

Waste system optimization should consider the most efficient means of utilization of these funds and appropriate divisions of responsibility between Government, utilities and private industry for the benefit of the rate payer and the public in general.

DOE is interested in identifying the most efficient waste management system that is practical and that fulfills the intent of the NWPA. We do not propose to have all of the answers at this time, nor do we feel that we have a corner in good ideas. Our desire is to work
closely with utilities and industry to identify the best ideas for the waste management system from a total systems point of view and then to study these ideas to see which ones offer the most advantages.

To accomplish this, DOE issued a competitive solicitation called a Program Research and Development Announcement (PRDA) in March of this year. The objective of this PRDA was to invite utilities and industry to identify and develop technical and institutional alternatives for near-term and long-term handling, packaging, shipping, and storing of spent fuel in ways that could facilitate and/or minimize handling and packaging for final disposal. Thus, the time and effort, safety and costs, throughout the back-end of the fuel cycle could be minimized while possibly providing near-term benefits to the utilities in solving storage problems in this process. Concepts for standardization to the maximum extent practical were to be considered that can lead to additional economies. Emphasis was placed on the waste packaging and handling system as a whole and how an improvement in one part of the system will affect the total system in terms of costs, safety and licensing.

The PRDA was issued on March 15, 1984 and proposals were due to DOE on May 15, 1984. Twenty-seven proposals were received from twenty-two proposers. Forty subcontractors, including utilities were included as participants in the proposals. The proposals covered a variety of ideas, some very unique, while others were for improvements to existing concepts.

Six proposals were selected by DOE for negotiation of awards. The successful proposers were:

**GA TECHNOLOGIES:** Proposed to study a multi-element system consisting of standard canisters and various overpacks or casks for storage and transportation.
E.R. JOHNSON: Proposed to study the advantages of a multi-purpose centralized packaging facility to serve the repository and/or MRS.

NUS: Proposed to study the advantages of utilizing a combination of square and half-square canister designs for spent fuel rods.

RIDIHALGH, EGERS AND ASSOCIATES: Proposed to further define and characterize the Minimax systems concept that features several regional packaging facilities.

TRANSNUCLEAR: Proposed to develop an extra large storage cask, a triangular-shaped canister and a compact mobile dry consolidation process.

WESTINGHOUSE: Proposed to study several metal canister and cask concepts, including the universal cask concept.

Contracts are currently being negotiated with the successful proposers. These studies are expected to take approximately twelve months. The most promising concepts will be pursued through further design development and field testing.

In addition to the PRDA studies, DOE has also initiated systems studies on the benefits to the whole waste system of rod consolidation and extended burnup of fuel. Other systems concepts will be studied as deemed appropriate.

While these systems studies are being conducted, existing design efforts in the various program areas (repository, MRS, transportation, etc.) will continue. When the results of the system studies are available, they will be evaluated along with the existing design data and the optimum waste management system will be defined and documented in a Systems Design Description (SDD) document. The SDD will then be used by the individual programs to redirect, if necessary, current design efforts. Figure 5 is a flow chart of this process.
We are aware that we are not the only ones looking at the waste management problem from a total systems concept. The Tennessee Valley Authority and the Electric Power Research Institute are also conducting systems studies on their own. We are anxious to co-ordinate our studies with other related studies.

In summary, our objective is to optimize the waste management system from the source of the fuel to the final disposition making maximum utilization of past and currently on-going work and incorporating new ideas as appropriate. We look forward to a continued working relationship with industry and the utilities to accomplish this task.
FIGURE 1
SPENT FUEL LOGISTICS

FEDERAL INTERIM STORAGE

AT-REACTOR STORAGE

REPROCESSING

REPOSITORY

MONITORED RETRIEVABLE STORAGE

OTHER

8/6/84
FIGURE 5
ILLUSTRATION OF THE SYSTEMS INTEGRATION PROGRAM LOGIC

REFERENCE CASE SDD

RESULTS OF SUPPLEMENTARY STUDIES

CURRENT REPOSITORY MRS. TRANSPORTATION DESIGN ACTIVITIES

RESULTS OF EVALUATION OF ALL INFORMATION TO DATE

PROPOSED MODIFICATIONS TO SDD

REVISED REFERENCE SDD

INTEGRATION INTO SUBSYSTEM DESIGNS (REPOSITORY, MRS. TRANSPORTATION)

RESULTS OF PRDA STUDIES

RESULTS OF ADDITIONAL SUPPLEMENTARY STUDIES AS NEEDED

9/11/84
THE FRENCH UTILITY'S VIEW ON IRRADIATED FUEL MANAGEMENT

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The purpose of this paper is to give some information about the philosophy and the experience of Electricité de France about the back end of the fuel cycle.

As indicated on the map (Figure 1), the present situation of the Electricité de France nuclear program is the following:

- **as PWR 900 MWE:** 28 units connected to the grid and six units on commissioning test or under construction
- **as PWR 1300 MWE:** two units connected to the grid and 19 units under construction

The units in operation have supplied 12,846 GWh during September 1984 and their average plant factor is 78.4 percent from the beginning of the year.

Furthermore, five gas-cooled graphite-moderated reactors are still in operation and Electricité de France is participating in France in four other nuclear units, among which are two fast breeder reactors. As a complement, the COGEMA reprocessing facility of La Hague is located in the northwestern part of the country, near the nuclear power plant of Flamanville.

With such a number of units, the whole nuclear fuel cycle has to be considered very carefully and in particular the back end.

Figure 2 shows the fuel cycle taking into account the three types of reactors which are managed by Electricité de France. The main steps of the cycle and the chemical form under which the nuclear material is used are represented. It can be noted that reprocessing is the common end of the chains; this is the constant philosophy of Electricité de France.
Today the loops using the thermal reactors are completed; as a matter of fact, the plutonium obtained after reprocessing of the fuel from these reactors was used for the fabrication of the first core of the fast breeder reactor Super Phenix for which Electricité de France has supplied more than three tons of plutonium. Then, the uranyl nitrate obtained after reprocessing of the first 180 fuel assemblies of Fessenheim and Bugey units has been sent to the DOE enrichment facility as feed material. In order to continue this loop, Electricité de France is studying with COGEMA and the French nuclear fuel manufacturer the possibility of fabricating a certain number of fuel assemblies using this re-enriched uranium.

With such a cycle, the storage of the spent fuel assemblies is only considered as an intermediate solution. After at least three stays in the reactor, the irradiated assemblies are unloaded and stored in the spent fuel pool of the fuel building. Figure 3 shows the cumulative amount of irradiated assemblies which will be unloaded up to the 90’s. Within two months after unloading, each assembly is systematically submitted to a sipping test in order to know its condition. At the beginning of this year we have detected 84 damaged assemblies, among which 68 presented some leaks. These damaged assemblies are stored without particular precautions but they cannot be handled before final evacuation. All the undamaged assemblies which have reached the burn-up forecast by their initial enrichment assay can be evacuated to the reprocessing facility; but, in general, 10 to 30 of them are kept in reserve to be reloaded once more in the reactor in case of necessity.

The storage capacity of the spent fuel pool of a PWR unit is six or seven thirds of the fuel in a reactor core. For safety reasons we are obliged to leave the place free for one core for a possible unloading of the reactor in case of necessity. This gives a capacity of three or four thirds of the fuel in a core for storage.
As far as the fast breeder reactor is concerned, a storage building is being built at Creys Malville site; after washing in the reactor building, the fuel assemblies could be stored under water for several years.

After a cooling period of at least one year, the PWR irradiated assemblies are transported by train to the La Hague reprocessing facility.

One of our main problems about the back end of the fuel cycle is to optimize spent fuel transport. Presently, we have transported to La Hague about 1400 assemblies with only seven casks containing 12 PWR assemblies each. In the future we will have to adjust the number of casks and to choose adequately the units from which the assemblies will be taken in order to minimize the transport cost, taking into account the increasing number of units in operation and the higher power of the new units, consequently the increasing number of fuel assemblies. But this increase will be moderated by an increase of the campaign length. The present nominal burn-up is about 33,000 MWD/t with an initial enrichment assay of 3.25 percent U235; the corresponding length of the campaign is approximately 11 months. During 1987, the new reload will be enriched at 3.7 percent giving about 36,000 MWD/t and 13 to 14 months of continuous operation of the reactor. The next step will be in the 90's with an enrichment level of about 4.2-4.3 percent and a campaign length of 16 to 18 months for a burn-up reaching 45,000 MWD/t. These changes will lead to a decrease in the spent fuel inventory in the units with respect to the amount forecast today, and it will be helpful for us in optimizing spent fuel transport.

If some damaged irradiated assemblies had to be transported, it would not be a big problem. If an assembly is very damaged, it could be evacuated alone within a specific cask, for example to a specialized laboratory of the French Commissariat a l'Energie Atomique; if it is only slightly damaged or leaking, it could be transported within a special box but, nevertheless, in a normal cask with other assemblies. Furthermore, the new design of the fuel assembly will allow a possible failed fuel rod
to be withdrawn. Such a rod will be replaced by a stainless steel rod and then will be placed into a special structure the size of which is the same as a fuel assembly. This structure could be transported as an assembly, if necessary within the special box.

After transport by train, the irradiated fuel assemblies are stored in the La Hague pools. Three or four years later, they are reprocessed following a program defined by Electricite de France. The storage at La Hague does not raise difficulties. As a matter of fact, as it can be seen on Figure 4, the investments developed by COGEMA lead to a very good agreement between the storage needs and the storage capacity. The curve A gives the number of irradiated assemblies which arrive at the La Hague facility from Electricite de France and from COGEMA foreign customers. The curve C gives the storage needs taking into account the reprocessing program as forecast by COGEMA. And the curve B shows the storage capacity of the pools already in operation (two pools) and under construction (two more pools).

The number of assemblies indicated by this curve B is calculated with the nominal configuration of storage, that is to say 600 baskets per pool with nine PWR assemblies each. But this capacity can easily be increased by reducing the pitch between the baskets or by using consolidated assemblies. Electricite de France is also interested by this last solution and we follow carefully the studies conducted on this subject.

The above text presents the Electricite de France's philosophy about the back end of the nuclear fuel cycle. It is based on several assumptions, mainly as far as the development of fast breeder reactors in France is concerned. But it is obvious that some of these assumptions could be revised in the future and we have thought of some alternative solutions but without bringing reprocessing into question again, except for very important changes. For example, if the fast breeder reactor development is slower than forecast, we could use plutonium in the PWR reactors; but in this case, we do not know if it will be economically
interesting to perform a second reprocessing operation; such a problem will arise with assemblies fabricated with reprocessed and re-enriched uranium.

This is the reason why we have begun to study a long term storage solution. Among the possible solutions, we think that the best one for Electricité de France is a large pool. As we have presently 20 nuclear sites, it can be possible to find a site having the best seismic and environmental conditions and in which we could build a pool for about 1000 assemblies. After 20 to 25 years, we think that dry storage will constitute the best solution. Although we have not yet studied such a storage type, we are interested in carefully following the development of cask and vault studies.

As a conclusion, it can be repeated that the Electricité de France's philosophy is to reprocess the irradiated fuel as soon as possible. If, under the pressure of important events, this philosophy had to be changed, we would be ready to revise our position. But, in this case, our intermediate storage capacity is large enough to allow us to choose the best solution in due time.
NUCLEAR FUEL CYCLE

GAS COOLED REACTORS

PWR

UF₆

ENRICHMENT

U

ELEMENT FABRICATION

REACTOR

AT-REACTOR STORAGE

STORAGE AT REPROCESSING FACILITY

REPROCESSING

PuO₂

FAST BREEDER REACTORS

PuO₂

PuO₂

ASSEMBLY FABRICATION

REACTOR

AT-REACTOR STORAGE

STORAGE AT REPROCESSING FACILITY

REPROCESSING

U N H

FIGURE 2
STORAGE NEEDS AND CAPACITY AT LA HAGUE

A - Number of PWR fuel assemblies arriving at La Hague

B - Storage capacity of La Hague pools

C - Storage needs with reprocessing
SESSION 2

WATER POOL STORAGE

Chairman: P.T. Truant
Ontario Hydro
ABSTRACT

The characteristics of Ontario Hydro's fuel and at-reactor irradiated fuel storage water pools (or irradiated fuel bays) are described. With on-power fuelling of reactors, each reactor of > 500 MW(e) net discharges an average of 10 or more irradiated fuel bundles to bay storage every full power day. The logistics of handling such large quantities of irradiated fuel bundles present a formidable challenge. The development of high density fuel storage containers and remote handling mechanisms and the use of several irradiated fuel bays at each reactor site have all contributed to the safe handling of the large quantities of irradiated fuel. Routine operation of the irradiated fuel bays over a period of more than 20 years and some unusual events in the bay operation are described.

It is concluded that the operation of Ontario Hydro's irradiated fuel storage bays has been relatively trouble-free despite the large quantity of fuel involved, and wet storage provides safe, reliable storage of irradiated fuel. Evidence indicates that there will be no significant change in irradiated fuel integrity over a 50 year wet storage period.

*Presented by R.T. Lee (Ontario Hydro) at the Workshop 53290
1.0 INTRODUCTION

Ontario Hydro presently operates 13 CANDU-PHW* nuclear reactors with a total capacity of more than 7000 MW(e) net. Details of these and other Ontario Hydro reactor units being commissioned or under construction are given in Table 1. The CANDU-PHW units, which are fuelled at power, use natural UO$_2$ fuel. With on power fuelling, a typical four unit station discharges 40 to 50 irradiated fuel (IF) bundles during a full power day. Such large quantities of irradiated fuel bundles provide station operating staff with inherent challenges in dealing with irradiated fuel handling and management.

Based on the excellent Atomic Energy of Canada Limited (AECL) experience with waterpool storage of IF, dating from 1947 (Remington et al, 1983), Ontario Hydro has used water pools (or irradiated fuel bays as they are called in Ontario Hydro) for IF storage at all its reactor sites (Table 1).

The zirconium alloy clad natural uranium dioxide CANDU fuel has proven to be ideally suited for wet storage. The fuel bundles, (Figure 1), are about 50 cm long by about 10 cm in diameter and weigh about 25 kg each. At an average reactor discharge burnup of 650 GJ/kg U (180 MWh/kgU), the fuel contains about 0.22 percent $^{235}$U and 0.38 percent total plutonium (0.28 percent fissile Pu), has a decay heat output of less than 1 to 5 kW/bundle after one day's cooling, and will not go critical in light water storage regardless of storage density or age of the fuel. With these characteristics, the irradiated fuel bundles can be closely packed in simple storage containers stacked on the irradiated fuel bay (IFB) floor.

* CANDU = Canadian Deuterium Uranium Reactor
  PHW = Pressurized Heavy Water
Ontario Hydro's irradiated fuel bays presently store over 300,000 bundles (Table 2). By the year 2000, this figure will increase by a factor of more than five.

This paper describes:

1. The characteristics of CANDU irradiated fuel, and Ontario Hydro irradiated fuel bays (IFB's).

2. The successful routine operation and performance of the IFB's, and how the logistics of handling the large numbers of fuel bundles involved have been successfully resolved.

3. Some of the operating problems experienced.

4. An on-going program to evaluate the long-term integrity of irradiated fuel in IFB storage.

2.0. CHARACTERISTICS OF ONTARIO HYDRO FUEL AND IRRADIATED FUEL BAYS

2.1 Fuel Characteristics

From the beginning of the Canadian nuclear program, the objective has been to develop power-reactor fuels that are reliable, inexpensive, and have low parasitic neutron absorption. To achieve this objective, the CANDU fuel design has been kept simple, as shown in Figure 1. Fabrication techniques are also simple, and, where possible, adapted from normal industrial practice. These techniques lend themselves to standardization and automation, thus minimizing the number of different fabrication processes.
Dimensions and other characteristics of Ontario Hydro's fuel are given in Table 3.

All Ontario Hydro fuel bundles fabricated since 1974 have a thin graphite layer (called Canlub) up to 20 μm thick on the inside surface of the zircaloxy-4 clad. This Canlub layer, which reduces susceptibility to stress corrosion cracking on reactor power ramps, has led to a low in-reactor defect rate, (i.e. <0.1 percent).

2.2 On-Site Irradiated Fuel Bays (IFB's)

2.2.1 General Description

Data on the type, liner material, size, fuel capacity and estimated fill date for the IFB's at Ontario Hydro's nuclear generating stations (NGS) are given in Table 4. The earliest stations, NPD* and Douglas Point*, had sufficient IFB storage capacity for the station life. The other stations (Pickering A, Pickering B, Bruce A and Bruce B) will need additional storage capacity beyond existing IFB's starting in the mid-1990's; Darlington will also need additional IF storage capacity in 1996. This paper will focus mainly on the Pickering and Bruce sites, as they alone account for over 90% of all irradiated fuel presently stored at Ontario Hydro's stations. The on-site IFBs are of two types:

1. Primary bays (PIFBs).
2. Auxiliary or secondary bays (AIFBs).

Irradiated fuel is discharged directly from Ontario Hydro's reactors to the primary irradiated fuel bays for initial storage and cooling. The primary IFBs consist of two compartments, separated by a hydraulically operated gate. The two compartments are:

* NPD and Douglas Point NGS are owned by AECL but operated by Ontario Hydro.
1. The receiving bay to which IF is discharged from the reactor directly. In this bay the IF is stacked in storage containers (Figure 2), possibly inspected, and later transferred to the second storage compartment known as the storage bay. There are facilities for canning defected IF, if required (see Section 3.4 below).

2. The storage bay, where the IF is stored in stainless steel storage containers called baskets, trays or modules (Figure 2). The receiving and storage bays generally have separate cooling and purification systems.

The basket is the container used to initially store irradiated fuel bundles in the Pickering A and Pickering B PIFB's. The tray is used to stack IF bundles in the Bruce A and Bruce B PIFB's (and the Bruce A AIFB). The module is a newer container designed to store the IF at about 1.5 times the storage density in the IFB compared to baskets i.e., 2189 kg U/m$^3$ (for the module) and 1393 kg U/m$^3$ (for the basket). The module not only provides for a higher storage density but has also been designed as an IF container for irradiated fuel transportation, which reduces double-handling of the bundles. Thus, all Pickering A and B IF bundles will eventually be transferred from baskets to module storage to optimize the IFB storage capacity.

The AIFBs, consisting of a single compartment, are very similar to the PIFBs in function and operation. They are designed to receive and store fuel after its initial cooling in PIFBs, and provide additional storage capacity as needed. The AIFB's also have provision for receiving IF transportation casks. Because of the reduced radioactivity of IF bundles when transferred to the AIFB's, the bundles need less water shielding. Thus in the AIFB's, the IF can be stacked closer to the water surface.
The IFB walls and floor are steel-reinforced concrete about two metres thick, and are either in-ground or above-ground structures. All inner IFB walls and floors are lined with either stainless steel or a fibreglass-reinforced epoxy compound, to form a watertight liner.

In all the bays, water is circulated through cooling and purification circuits, which are described in Section 2.2.2 below. Methods used to control water purity are a combination of ion exchange columns, filters and skimmers.

Ontario Hydro's IFB's use various liners and water purification systems. The choice of these components has been made on the basis of economics for the particular nuclear generating station concerned.

2.2.2 Cooling and Purification Systems

(a) Cooling

Cooling of bay water is achieved by tube and shell heat exchangers, with demineralized IFB water on the tube side and raw lake or river water on the shell side.

As the irradiated fuel in the AIFB's has been stored for at least three months in the PIFBs, the AIFB cooling system capacity is proportionally smaller than that needed for the PIFBs.

(b) Purification

All IFB purification systems are designed to remove suspended and dissolved solids (both of which may be radioactive). The IFB purification system components and flow capacity for Pickering A and B, Bruce A and B and Darlington are shown in Table 5.
In addition, water flows continuously through skimmers located at the water surface at intervals around the bay walls to remove any floating solids. Vacuum system type equipment is used at a frequency of once every 2 or more years to remove solids deposited on the bay floor and ledges.

The AIFB purification system capacity in general is proportionally less than that of the PIFB purification system, because any leaching of radioisotopes from clad crud and defected fuel is at a reduced rate.

2.2.3 Irradiated Fuel Bay Water Specifications

(a) Chemical Specifications

Chemical control is maintained:

1. To minimize corrosion of metal surfaces, e.g. fuel clad, stainless steel bay liner, storage containers, stacking frames, and handling tools,

2. To minimize the level of radioisotopes in the water, and as a result reduce the radiation fields and radioiodine levels in the bay area, and

3. To maintain clarity of the bay water for ease of bay operation.

The water purity is maintained by using only demineralized make-up water and close chemistry control based on pH, conductivity and for the Pickering and Bruce bays, chloride concentration (specified values are quoted in Table 6).
(b) Temperature Specifications

The temperature of the bay water is maintained at $32^\circ C$.

The temperature specification has been selected to prevent excessive stresses in the bay walls which could eventually lead to cracking of the concrete. Such a bay water specification also provides comfortable working conditions (i.e. air temperature and humidity) for personnel in the IFB vicinity.

3.0 ROUTINE OPERATION

3.1 Background

The early operating experience gained at NPD and Douglas Point stations has provided a basis for the successful operation of the irradiated fuel bays at Pickering and Bruce sites. The early experience and the development of high density storage containers (see Section 2.2.1), interbay fuel transfers, and remote handling mechanisms have all contributed towards meeting the logistics challenge of handling large quantities of IF bundles (see Section 1) in an economical and safe manner.

3.2 Irradiated Fuel Handling and Storage

3.2.1 Primary Bay

The fuel arrives underwater at the receiving bay of the primary bay, in pairs by conveyor (Pickering Nuclear Generating Station-A or PNGS-A) or by a port (Bruce Nuclear Generating Station-A or BNGS-A) mounted with a discharge mechanism.
At PNGS-A, each pair of bundles is pushed via a ram into a basket. Once the basket is full, the bay gantry crane moves the basket to the storage area of the bay where it is stacked vertically on stacking frames (Figure 3) no more than six baskets high in order to maintain an effective water barrier for shielding. These stacking frames maintain a clearance of 45 cm between the bottom of the filled baskets and the floor to ensure that the flow of cooling water is uninterrupted and that the epoxy liner has adequate water shielding for radiation protection.

Baskets are loaded in a similar fashion in the PNGS-B primary bay. However, once the baskets are filled, the bundles are transferred from baskets to the higher density module containers. The modules are then placed onto a stacking frame six modules high.

At BNGS-A, the discharge mechanism lowers each pair of bundles onto racks which are placed on an indexing mechanism located below the irradiated fuel discharge port. The crane operator transfers the bundles from the racks onto the storage trays. The full trays are then moved to the storage section of the bay where they are stacked 15 high.*

3.2.2 Irradiated Fuel Transfer to Auxiliary Bay

As stated in Section 2.2.1, the auxiliary bay provides an interim storage facility to handle the irradiated fuel volume which is in excess of the capacity of the primary irradiated fuel bay. Figure 4 depicts diagrammatically the fuel handling/storage/transfer in the PNGS-A/B and BNGS-A bays.

* Because the stainless steel floor liner is more tolerant to radiation than an epoxy liner, a clearance of only 18 cm is maintained between the fuel trays and the floor at Bruce NGS.
The PNGS-A primary bay is connected to the auxiliary bay by an enclosed corridor. For each irradiated fuel transfer operation, eight baskets of at least 4-year old irradiated fuel are selected from the PNGS-A primary bay and loaded underwater into the on-site cask. The bundle age restriction ensures acceptable radiation fields from the on-site shipping cask during transfer operations. After washing down, the cask is loaded onto the transfer vehicle (Figure 5) and moved through the enclosed corridor (~200 m distance) to the auxiliary bay. The maximum rate of travel for the transfer vehicle is 0.25 m/s. Once the cask is lowered into the auxiliary bay and unloaded, a basket-to-module transfer is carried out. The modules are then stacked seven high. PNGS-B does not have an auxiliary bay. Various options to provide additional storage space at the Pickering site are presently being evaluated.

At BNGS-A, the trays of irradiated fuel remain in the primary bay for a minimum of 3 months. Approximately every 4 months, roughly 300 trays are transferred to the auxiliary bay. Trays are transferred two at a time on a cart which travels through a water-filled tunnel connecting the two bays. The trays are then stored in stacking frames. A program is currently underway to increase the storage capacity of the auxiliary bay by approximately 3.5 station years arisings of fuel. This involves the installation of new stacking frames which allow for closer spacing coupled with higher stacks, i.e. 37 trays high compared with 32 trays high with the previous configuration.

3.3 Cooling and Purification

The normal operating temperature of the PNGS-A primary bay is 23 to 32°C with two heat exchangers on-line. If the temperature exceeds the specified maximum value of 32°C, a third heat exchanger is valved in. Some fouling of the shell side of the Pickering and Bruce PIFB heat exchanger tubing has necessitated
periodic chemical cleaning of the heat exchangers to restore their cooling capability. The frequency of cleaning is from one to five years. Section 4.1 details the heat exchanger fouling problems and chemical cleaning procedures at both stations.

Good chemical control has been achieved in both PNGS-A and BNGS-A irradiated fuel bays. A survey covering the 1978 to 1982 period indicated that the pH, Cl⁻, and conductivity levels have remained within specification most of the time, the only exception being a single conductivity measurement made in the PNGS-A PIFB which was 30% higher than specified. A high conductivity reading on the outlet of an ion exchange column indicates when the resin is spent. For the Pickering and Bruce PIFB's, this occurs about once per year.

With this close chemical control, the effect of bay water contamination on the long-term integrity of IF clad and other bay metal surfaces is considered to be insignificant.

3.4 Handling of Defected Irradiated Fuel

Since the CANLUB fuel design has been in use, (i.e. since 1974, see Section 2.1), the overall PNGS-A and BNGS-A IF defect rate has been low, i.e. <0.1% (a total of 221 bundles have defected). During early operation, the canning (i.e. the storing of a bundle in a sealed cylinder) of defected fuel was carried out. As more operating experience was gained, canning of defected fuel has become a contingency rather than a routine operation, due to the minimal release of fission products from most defected bundles.

At Bruce NGS-A, an on-power defect detection system serves to identify reactor fuel channels containing defected fuel. Once identified, fuel from such a channel is removed at the earliest possible date. Each bundle pair is pushed into the discharge mechanism and kept there while air from the mechanism cavity is purged past a gamma detector to identify the defect bundles. The suspect
bundles are then transferred to a tray in the normal fashion. This tray is segregated from the rest of the irradiated fuel until the suspect bundles on it can be inspected. After inspection, all intact bundles are returned to normal storage. Defected bundles are stored in a special location in the bay and, depending upon the severity of the defect, some may be canned.

Pickering-A PIFB has IF canning facilities but with the excellent fuel performance, no IF bundles have been canned since 1974.

The plan for PNGS-B is not to send any known defected fuel to the IFB until the defected bundle has had 2 to 3 days to cool and allow fission products to decay while held temporarily in the fuel handling systems.

4.0 UNUSUAL EVENTS

4.1 Background

In light of the excellent overall performance of underwater IF storage, operational problems experienced at the IFB's have been minimal. Two unusual events which have occurred are described below.

4.2 Heat Exchanger Tube Fouling and Cleaning

(a) Pickering NGS-A

During the summer months of each year from 1975, it has been difficult to provide sufficient cooling of primary bay water to maintain the temperature in the 23°C to 32°C operating range, even with two heat exchangers, HX1 and HX2, on line. If the bay temperature exceeds 32°C on a regular basis, there is a risk of minor damage to the concrete walls (see Section 2.2.3). In 1979, a third heat exchanger was installed to allow inspection of HX1 and
HX2. The latter were both found to be seriously fouled. Chemical cleaning of HX1 and HX2 with 10% formic acid resulted in the removal of 50 kg of deposit from each heat exchanger (the tubing area is about 365 m² per heat exchanger). The deposit fouling the heat exchangers on the shell side was a mixture of calcium carbonate, iron oxides and silica, with an approximate thickness of 1.5 mm. However, a post-cleaning inspection revealed that although the straight legs of the tube bundle were effectively cleaned, the U-bend region was not.

Formic acid cleaning was used again during 1981 and 1982 with similar results. Although most of the calcium carbonate was removed, silt and mud deposits still remained in the U-bend region. Laboratory tests to identify a more effective cleaning solvent resulted in a recommendation to use ammoniated citric acid solution. This method will be incorporated in the next heat exchanger cleaning operation.

(b) **Bruce NGS-A**

In late June 1980, the primary bay water temperature rose to approximately 37°C. With HX (heat exchanger) 2 and HX3 operating with maximum cooling water flow, HX1 was valved in to cool the PIFB back to below 32°C. Fibre optics inspection of the shell side (raw lake water) of HX3 showed the tube nest to be solidly blocked with deposits. The composition of the deposits consisted of calcium carbonate, iron oxide, and silica. HX1 was found to be similarly fouled.

In October 1980, HX3 and HX1 were chemically cleaned with inhibited 10% formic acid followed by a neutralizing solution. After cleaning, the tubes were visually inspected revealing that only a very thin deposit remained. Thus the cleaning method was successful. A total of 140 kg of calcium carbonate and 63 kg of iron compounds were removed from the two heat exchangers (the shell side tube area is about 555 m² per heat exchanger). The tube surface of HX2 was inspected and found to be clean.
Presently, there has been no further need to repeat the cleaning. However, a program to routinely monitor the cooling capability of the heat exchangers has been implemented.

4.3 Use of Hydrazine to Reduce Volatile Iodine Levels

During 1972, PNGS A experienced a high fuel defect rate caused by the initial fuel management scheme. Upon discharge to the primary IFB, the defected fuel released sufficient quantities of iodine to generate high airborne iodine activity. There were no incidents of high radiation exposure of personnel. Tests indicated that the addition of hydrazine to the IFB water reduced the oxidized forms of radioiodine and led to a significant reduction in airborne iodine activity. It was also observed that hydrazine effectively reduced the release of radioiodine under transient conditions when fresh defected fuel bundles were discharged into the bay.

Actual tests conducted in IFB water indicated that a decrease of airborne I-131 activity by a factor greater than seven was observed 15 minutes after hydrazine was added to the receiving bay (to give 125 mg/kg hydrazine) and to the storage bay water (to give 5 mg/kg hydrazine). It was also confirmed that hydrazine is not rapidly decomposed by atmospheric and dissolved oxygen at the temperature and chemical conditions in the bay water. It took about 48 hours for almost all the hydrazine in the bay water to be decomposed. Hydrazine also has an advantage over many other chemicals in that its main reaction with oxygen results in the formation of water and nitrogen which do not affect bay operation. To avoid eluting any ions from the IFB purification system ion exchange columns, the latter are valved out prior to hydrazine addition and not valved in until the hydrazine concentration falls to 1 mg/kg.

It has not been necessary to use hydrazine addition to the PNGS A PIFB since 1972 due to the excellent reliability of the irradiated fuel.
5.0 LONG-TERM IRRADIATED FUEL INTEGRITY IN WET STORAGE

5.1 Background

A key element in irradiated fuel management is to ensure the IF integrity during the various phases of its handling and management, including IFB storage. Thus, Ontario Hydro and AECL have a program (Hunt et al, 1981), initiated in 1977, to examine irradiated fuel stored in IFBs for possible deterioration.

Nineteen bundles from the Douglas Point, Pickering and NPND generating stations and the AECL Chalk River NRU prototype reactor are being examined. The oldest bundles have been in wet storage since 1962.

Seven destructive and non-destructive tests have been selected to characterize the elements initially and in subsequent re-examinations after further wet storage periods. The tests used to determine if there is any deterioration of either the uranium dioxide fuel (with defected cladding) or the Zircaloy cladding, are as follows:

1. Neutron radiography.

2. Fission gas analysis.

3. Hydrogen and deuterium analysis.

4. Ring tensile tests.

5. Visual examination.


7. Torque tests.
Post-irradiation (i.e. from the time when they were first discharged from the reactor) data from such tests is available for many of the bundles for comparison with recent examination results.

The original re-examination period was every five years starting in 1978. However, since no IF deterioration was detected (Hunt et al, 1981), this period has been increased to ten years. The second set of re-examination tests is thus scheduled for 1988, and the final set for 2008. This will give results on any fuel deterioration in wet storage up to about 45 years. All seven of the tests described above will be repeated for each re-examination.

5.2 Results and Discussion

The results (Hunt et al, 1981) of the characterization tests and the first set of re-examination tests show no apparent irradiated fuel deterioration of either the uranium dioxide fuel matrix (for 'affected fuel) or Zircaloy cladding due to storage in IFB's for a time period up to 17 years. Based on results to date, irradiated fuel should maintain its integrity during fifty years of underwater IFB storage. With future characterization results, this predicted period may be extended.

6.0 CONCLUSIONS

Ontario Hydro has gained considerable experience in the design, construction and operation of irradiated fuel storage facilities. Water-filled bays at the reactor sites have been designed with capacities ranging from about 700 Mg to 7,000 Mg of irradiated fuel. Auxiliary irradiated fuel bay storage facilities have also been constructed at the reactor sites. Irradiated fuel is being successfully transferred from the primary storage bays to these auxiliary bays by means of on-site flask/vehicle systems and conveyor systems.
A new irradiated fuel storage container, the module, has been designed to provide a higher density fuel bay storage. The module has also been designed as the irradiated fuel container for off-site transportation, thus minimizing fuel handling operations at the storage/transportation interface.

Routine operation over a period of more than 20 years of the Ontario Hydro-operated irradiated fuel bays has been relatively trouble-free, and the bays have provided safe, reliable interim storage of irradiated fuel bundles.

Tests on irradiated fuel after wet storage for periods up to 17 years indicate no fuel deterioration, whether it is defected (i.e. with a through-wall defect in the clad) or not. All evidence to date suggests there will be no significant change in irradiated fuel bundle integrity over a 50 year wet storage period, whether or not there are any fuel clad through-wall defects.

REFERENCES


ACKNOWLEDGEMENTS

Thanks are due to R.T. Lee of Ontario Hydro's Central Nuclear Services who provided most of the IFB operations-related material in this paper. The help of NPD, Douglas Point, Pickering, Bruce (particularly I. McIntyre) and Darlington station and design staff for providing additional operational and other background information is also acknowledged.
TABLE 1:
ONTARIO HYDRO'S NUCLEAR GENERATING STATIONS*

<table>
<thead>
<tr>
<th>Station</th>
<th>Capacity/unit (MW(e) net)</th>
<th>No. of Units</th>
<th>In-Service Date</th>
<th>First Unit</th>
<th>Last Unit</th>
</tr>
</thead>
<tbody>
<tr>
<td>NPD</td>
<td>22</td>
<td>1</td>
<td>1962</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Douglas Point</td>
<td>206</td>
<td>1</td>
<td>1968</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Pickering A</td>
<td>515</td>
<td>4</td>
<td>1971</td>
<td>1973</td>
<td></td>
</tr>
<tr>
<td>Pickering B**</td>
<td>516</td>
<td>4</td>
<td>1983</td>
<td>1985</td>
<td></td>
</tr>
<tr>
<td>Bruce A</td>
<td>740</td>
<td>4</td>
<td>1977</td>
<td>1979</td>
<td></td>
</tr>
<tr>
<td>Bruce B***</td>
<td>784</td>
<td>4</td>
<td>1984</td>
<td>1991</td>
<td></td>
</tr>
<tr>
<td>Darlington****</td>
<td>881</td>
<td>4</td>
<td>1988</td>
<td>1992</td>
<td></td>
</tr>
</tbody>
</table>

* All units are CANDU, with a pressurized heavy water coolant (or CANDU PHW)

** Two units are in-service, the others are being commissioned

*** One unit is in-service, the others are being commissioned

**** All units are under construction
### TABLE 2

**IRRADIATED FUEL QUANTITIES**  
(In Thousand Bundles)

<table>
<thead>
<tr>
<th>Nuclear Generating Station</th>
<th>Typical Annual IF Arisings</th>
<th>Bay</th>
<th>Inventory as of January 1, 1984</th>
</tr>
</thead>
<tbody>
<tr>
<td>NPD</td>
<td>0.25</td>
<td>Primary</td>
<td>4.0</td>
</tr>
<tr>
<td>Douglas Point</td>
<td>1.6</td>
<td>Primary</td>
<td>19.6</td>
</tr>
<tr>
<td>Pickering A</td>
<td>12.5</td>
<td>Primary</td>
<td>71.4</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Auxiliary</td>
<td>82.4</td>
</tr>
<tr>
<td>Pickering B</td>
<td>14.0</td>
<td>Primary</td>
<td>12.7</td>
</tr>
<tr>
<td>(Projected)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Bruce A</td>
<td>23.0</td>
<td>Primary</td>
<td>12.7</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Auxiliary</td>
<td>113.5</td>
</tr>
<tr>
<td>Bruce B</td>
<td>23.5</td>
<td>Primary</td>
<td>0.0</td>
</tr>
<tr>
<td>(Projected)</td>
<td></td>
<td>Auxiliary</td>
<td>0.0</td>
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<tr>
<td><strong>Total</strong></td>
<td><strong>74.8</strong></td>
<td></td>
<td><strong>Total</strong></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td><strong>305.5</strong></td>
</tr>
<tr>
<td>Reactor</td>
<td>NPD</td>
<td>Douglas</td>
<td>Pickering A</td>
</tr>
<tr>
<td>-------------</td>
<td>-----</td>
<td>---------</td>
<td>-------------</td>
</tr>
<tr>
<td>Number of Elements Per Bundle</td>
<td>19</td>
<td>19</td>
<td>28</td>
</tr>
</tbody>
</table>

**ELEMENTS**

<table>
<thead>
<tr>
<th>Material</th>
<th>ZIRC-4</th>
<th>ZIRC-4</th>
<th>ZIRC-4</th>
<th>ZIRC-4</th>
<th>ZIRC-4</th>
<th>ZIRC-4</th>
<th>ZIRC-4</th>
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</thead>
<tbody>
<tr>
<td>Outside Diameter mm</td>
<td>15.25</td>
<td>15.22</td>
<td>15.19</td>
<td>15.19</td>
<td>13.08</td>
<td>13.08</td>
<td>13.08</td>
</tr>
<tr>
<td>Min Cladding Thickness mm</td>
<td>0.38</td>
<td>0.38</td>
<td>0.38</td>
<td>0.38</td>
<td>0.38</td>
<td>0.38</td>
<td>0.38</td>
</tr>
</tbody>
</table>

**BUNDLES**

| Length mm | 495.3  | 495.3  | 495.3  | 495.3  | 495.3  | 495.3  | 495.3  |
| Maximum Diameter mm | 82.04  | 81.74  | 102.49 | 102.49 | 102.49 | 102.49 | 102.49 |

| Avg Discharge Bundle Burnup MWh/kgU | 188 | 199 | 203 | 193 | 201 | 178 | 180 |

* First unit is scheduled to go in-service in 1988
<table>
<thead>
<tr>
<th>Station</th>
<th>Type</th>
<th>Dimensions** (m)</th>
<th>Capacity 000's of Bundles</th>
<th>In-Service Date</th>
<th>Bay Fill* Date</th>
<th>Liner Material</th>
</tr>
</thead>
<tbody>
<tr>
<td>NPD</td>
<td>PIFB</td>
<td>4.3Wx7.3Lx5.5D</td>
<td>2</td>
<td>1962</td>
<td>***</td>
<td>All stainless steel (S/S)</td>
</tr>
<tr>
<td>Douglas Point****</td>
<td>PIFB(a)</td>
<td>3.4Wx7.3Lx7.2D</td>
<td>50</td>
<td>1966</td>
<td>****</td>
<td>All stainless steel (S/S)</td>
</tr>
<tr>
<td></td>
<td>PIFB(b)</td>
<td>7.6Wx20.9Lx7.2D</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pickering A</td>
<td>PIFB****</td>
<td>16.3Wx29.3Lx8.1D</td>
<td>93</td>
<td>1972</td>
<td>1994</td>
<td>All epoxy</td>
</tr>
<tr>
<td></td>
<td>AIFB</td>
<td>17Wx34Lx8.1D</td>
<td>214</td>
<td>1978</td>
<td>1994</td>
<td>All epoxy</td>
</tr>
<tr>
<td>Pickering B</td>
<td>PIFB</td>
<td>16.3Wx29.3Lx8.1D</td>
<td>158</td>
<td>1983</td>
<td>1995</td>
<td>Receiving bay - all S/S Storage bay, all epoxy</td>
</tr>
<tr>
<td>Bruce A</td>
<td>PIFB</td>
<td>10Wx41Lx6D</td>
<td>21</td>
<td>1977</td>
<td>1994</td>
<td>S/S floor, epoxy walls</td>
</tr>
<tr>
<td></td>
<td>AIFB</td>
<td>18Wx46Lx9D</td>
<td>352</td>
<td>1979</td>
<td>1994</td>
<td>S/S floor, epoxy walls</td>
</tr>
<tr>
<td>Bruce B</td>
<td>PIFB</td>
<td>10Wx46Lx6D</td>
<td>36</td>
<td>1983</td>
<td>2002</td>
<td>All S/S</td>
</tr>
<tr>
<td></td>
<td>AIFB</td>
<td>18Wx46Lx9D</td>
<td>330</td>
<td>1987</td>
<td>2002</td>
<td>All S/S</td>
</tr>
<tr>
<td>Darlington*****</td>
<td>PIFB</td>
<td>9.65Wx20.6Lx5D</td>
<td>212</td>
<td>1987</td>
<td>1996</td>
<td>All S/S</td>
</tr>
<tr>
<td></td>
<td>(a)</td>
<td>17Wx32Lx9.2D</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>(b)</td>
<td>17Wx4Lx9.2D</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

* Based on combined capacity of all bays on-site.

** W = width, L = length, D = depth

*** Irradiated fuel is transported to AECL/CRNL for storage after six months cooling at NPD

**** PIFB consists of an IF receiving bay (a) and an IF storage bay (b). As unit was shut down in 1984, the PIFB will never fill.

***** Based on storage using baskets. Transfer of bundles to higher density module storage would increase the capacity and extend the bay fill date to the year 2000.

****** Darlington will have two identical PIFB's, the second (east) one will be in-service in 1991, with the fill date about 2000. Each PIFB consists of an IF receiving bay(a), an IF storage bay (b) and an IF cask handling bay (c).
### TABLE 5
#### IRRADIATED FUEL BAY PURIFICATION SYSTEM CAPACITY

<table>
<thead>
<tr>
<th>Station</th>
<th>Bay</th>
<th>Purification Flowrate (L/s)</th>
<th>Purification Equipment</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pickering A</td>
<td>PIFB (receiving bay)</td>
<td>12.1</td>
<td>Ion exchange (IX) columns</td>
</tr>
<tr>
<td></td>
<td>PIFB (storage bay)</td>
<td>30.3</td>
<td>Ion exchange (IX) columns</td>
</tr>
<tr>
<td>Pickering A</td>
<td>AIFB</td>
<td>65</td>
<td>Filters, IX columns</td>
</tr>
<tr>
<td>Pickering B</td>
<td>PIFB (receiving bay)</td>
<td>63.7*</td>
<td>Filters, IX columns</td>
</tr>
<tr>
<td></td>
<td>PIFB (storage bay)</td>
<td>37.8</td>
<td>Filters, IX columns</td>
</tr>
<tr>
<td>Bruce A</td>
<td>PIFB (receiving bay)</td>
<td>75.6</td>
<td>IX columns</td>
</tr>
<tr>
<td></td>
<td>PIFB (storage bay)</td>
<td>37.8</td>
<td>IX columns</td>
</tr>
<tr>
<td>Bruce A</td>
<td>AIFB</td>
<td>37.8</td>
<td>IX columns</td>
</tr>
<tr>
<td>Bruce B</td>
<td>PIFB (receiving bay)</td>
<td>75.7</td>
<td>IX columns</td>
</tr>
<tr>
<td></td>
<td>(storage bay)</td>
<td>37.8</td>
<td>IX columns</td>
</tr>
<tr>
<td></td>
<td>AIFB</td>
<td>19</td>
<td>IX columns</td>
</tr>
<tr>
<td>Darlington</td>
<td>PIFB (receiving bay)</td>
<td>92</td>
<td>Filters, IX columns</td>
</tr>
<tr>
<td></td>
<td>PIFB (storage bay)</td>
<td>78</td>
<td>Filters, IX columns</td>
</tr>
<tr>
<td></td>
<td>(cask handling bay)</td>
<td>39</td>
<td>Filters, IX columns</td>
</tr>
</tbody>
</table>

* The purification system draws water from both the receiving bays and the storage bay together.
### TABLE 6
**IFB CHEMICAL CONTROL SPECIFICATIONS**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Specified Range</th>
</tr>
</thead>
<tbody>
<tr>
<td>pH</td>
<td>(For NPD, Douglas Point, Bruce A,B) (For Pickering A, B)</td>
</tr>
<tr>
<td></td>
<td>5.5 to 8.0</td>
</tr>
<tr>
<td>Chloride</td>
<td>$0.3 \text{ mg/kg*}$</td>
</tr>
<tr>
<td>Conductivity</td>
<td>$0.2 \text{ mS/m (or 2 \text{ \mu mho/cm})}$</td>
</tr>
</tbody>
</table>

* No chloride specification for NPD and Douglas Point.
FIGURE 1
Fuel Bundle for Pickering Reactor
Assembled from Seven Basic Components
FIGURE 2
Ontario Hydro Irradiated Fuel Storage Containers
FIGURE 3
Pickering NGS-A Irradiated Fuel Stacking Frame
FIGURE 4
Fuel Handling/Storage

Pickering NGS-A

PRIMARY BAY

- Baskets
- 4-Year Residency

TRACTOR/ON-SITE CASK CORRIDOR

AUXILIARY BAY

- Basket-to-Module Transfer

Pickering NGS-B

PRIMARY BAY

- Baskets
- Basket-to-Module Transfer

Bruce NGS-A

PRIMARY BAY

- Trays
- 4-Month Residency

UNDERWATER DUCT AND CONVEYOR

AUXILIARY BAY

- Trays
FIGURE 5
On-Site Flask Transfer Vehicle
WATER POOL OPERATIONAL EXPERIENCE AT MORRIS

E.E. Voiland
J.W. Doman
K.J. Eger

General Electric Company
Morris Operation
7555 E. Collins Road
Morris, Illinois 60550

ABSTRACT

At General Electric's Morris Operation, a spent nuclear fuel storage facility, a number of "studies" have been conducted in the last 12 years related to the storage of spent nuclear fuel, the IF-300 spent fuel transportation system, and various plant systems. A list of 30 such studies - some extensive and some not very extensive - is presented. The results of several study topics are also presented: hydrogen gas concentration in spent fuel transportation casks; correlation of surface dose rates with storage pool contaminant concentrations; heat generation rates of stored spent fuel; individual fuel bundle heat generation rates; and removal of radiocesium from a spent fuel storage pool by synthetic zeolites. Frequently, the results of such studies that are performed at operational facilities are not very well publicized. However, the experience of operating facilities may be a worthwhile source of practical information.
1. INTRODUCTION

In the operation of almost any facility, even one as "non-technical" as a spent nuclear fuel storage facility, various changes suggest themselves. Often changes in a nuclear facility are stimulated by the need to control personnel radiation exposure; others are stimulated by more conventional driving forces, such as improved operation, energy conservation, or saving money. More often than not these changes are evolutionary and are incorporated in the facility's operation without fanfare. Besides the stimulus of change, the highly regulated nature of the nuclear industry and the need to provide a basis for response to the (frequently speculative) concerns of nuclear critics provide additional reasons for "studies" yielding better information or understanding of technical aspects of our business.

General Electric's Morris Operation is the site of a spent nuclear fuel storage facility which has been operating for almost 13 years.* It also serves as the base for operation of three rail car mounted Type B transportation casks, each having a capacity of about 3.5 MTU.

The purpose of this presentation, is to provide an overview of the kinds of studies undertaken and facility changes made in the history of the site's operation. Some specific examples are presented which emphasize results rather than methodology.

* See Appendix A for a brief facility description.
2. SOME STUDY ACTIVITIES UNDERTAKEN AT MORRIS OPERATION

A partial list of tests and studies appears in Appendix B. The listed items range from rather extensive efforts (the measurement of thermal output of spent fuel, the measurement of radiation dose rates from spent fuel in air and water, and the experimental basis for using Zeolon-100* for radiocesium removal from storage pool water) to trivial measurements (hydrogen concentrations found in shipping casks full of spent fuel) or simple calculations (per-pound comparison of the heat output of standard man and spent fuel stored at Morris).

3. EXAMPLES OF RESULTS OF SOME STUDIES UNDERTAKEN AT MORRIS OPERATION

The following examples of studies undertaken at Morris Operation have been selected simply on our belief that they might be interesting to people concerned with spent fuel storage. Some of the studies have been well documented but, perhaps, not well publicized. Others were undertaken because of internal use or need and no external publication was made.

3.1 STORED FUEL HEAT GENERATION RATES

At various times between 1976 and the present, we have estimated the heat output of the stored fuel. The measured values were obtained by stopping the flow of basin water through the coolers and measuring the

* Registered Trademark of the Norton Co., Akron, Ohio.
rate of temperature rise in the storage basin over a several day interval. A correction of 44 kilowatts was added to compensate for water evaporation and convective losses to the ground. Figure 1 shows that the calculated values - using ANSI ANS 5.1 - are much higher than the measured values and suggested the desirability of more refined measurements on individual fuel assemblies for calculational code verification.

3.2 FUEL BUNDLE HEAT GENERATION RATES\(^{(1)}\)

As part of a US Department of Energy funded activity, Morris Operation developed a thermal output (power) measuring device consisting of an insulated chamber into which either an electrical heater or a spent fuel bundle could be inserted and the temperature rise in the water accurately measured. The thermal outputs of 15 pressurized water reactor bundles were measured using this device (Judson et al, 1982). Repeatability of the method was determined by repeated measurements on the same fuel assembly at various times during the program. Five tests of the same type disclosed values for the same bundle of 732, 726, 713, 719, and 731 watts for a mean value of 724 watts and a standard duration of \(\pm 8.1\) watts or slightly over one percent relative. Figure 2 shows plots of predicted power values versus measured values using various calculational methods. The curve through the origin represents perfect correlation. The lack of correspondence noted with use of the two ANSI methods and the ORIGEN (Constant Power) is quite evident.

The best fit occurred with use of the ORIGEN code and actual irradiation history - actual power levels, operating intervals data, etc. Only three points were calculated using this approach; however, the fit was excellent.
Later in 1984, the decay heat power of these same assemblies, was calculated (Schmittroth, 1984) and good correlation was found using actual reactor history for the input data. It is no surprise that using average power levels and assumed continuous reactor operation will bias calculated power values high, since these assumptions have the effect of artificially reducing the decay interval.

Currently, Morris Operation is using its capability to measure the decay heat power of 54 fuel assemblies as part of DOE's BWR Storage Cask Characterization Study, subject of another paper at this conference.

3.3 HYDROGEN GAS CONCENTRATION IN SPENT FUEL TRANSPORTATION CASKS
(N.P. Shaikh)

In 1976, Morris Operation cursorily examined the air void in four single element truck mounted shipping casks for the presence of hydrogen gas. The motivation was to determine whether concentrations within the explosive range might result during typical spent fuel shipments. It should be noted that the July 1976 measurements involved very short cooled fuel, about 150 days, and thus represent extreme radiation conditions. The hydrogen concentrations measured in the casks transporting high exposure, short cooled fuel are greater than those observed in the case of the long cooled material, but certainly do not linearly correlate with the perhaps tenfold more intense radiation field of the former. However, not much speculation is warranted by the sparse measurements of Table 1. Nevertheless, the data do provide some insight on hydrogen concentrations in a few wet shipments of spent fuel.
3.4 CORRELATION OF POOL SURFACE DOES RATES WITH RADIONUCLIDE CONCENTRATIONS

In 1979, some careful measurements of the dose rates (in millirem per hour) at surface of the storage pool were made. At the same time, concentrations of gamma ray emitting radionuclides in the water were measured. A correlation of expected dose rates calculated from radionuclide decay data showed a linear relationship but a one-to-one correspondence was lacking because build-up factors were (purposely) not included.

More recently, the 1979 data were re-examined and a semi-empirical approach was used to establish a relation between measured dose rates and water concentrations. In this approach, the concentrations of individual radionuclides are adjusted for the relative dose rates produced by identical curie-quantities of each nuclide. From tabulated values\(^3\) the relative dose rates from cobalt-60, cesium-134, and cesium-137 (the only gamma emitters of significance in Morris Operation pool water) are, respectively, 1, 0.66 and 0.25. When the actual concentrations are adjusted by these values, assumed to provide an overall equivalent concentration, and plotted versus the measured surface exposure, one obtains the plot of Figure 3. The empirical equation for this correlation is

\[
\text{Surface Exposure Rate (mR/hr)} = 0.23 \text{(Co-60)} + 0.25 \text{Cs-137} + 0.66 \text{Cs-134} + 0.9
\]

where Co-60, Cs-137, and Cs-134 are concentrations in units of \(10^{-4}\) microcuries per milliliter of the respective radionuclides.
It should be noted that this relationship is somewhat facility specific as the intercept is indicative of the dose rates from sources other than the pool water.

3.5 REMOVAL OF RADIOCESIUM FROM STORAGE POOL WATER BY ZEOLITES

During the period 1975-1976, laboratory investigations disclosed that a synthetic sodium aluminosilicate, Zeolon-100, had a very strong affinity for radiocesium in water. In this investigation, it was established - not surprisingly - that radiocesium behaved like an ionic species.

The affinity of cesium for the Zeolon-100 could then be quantified as a distribution coefficient.

\[
K_D = \frac{C_{\text{Ion Ex}}}{C_{\text{Sol}}}
\]

where \( C_{\text{Ion Ex}} \) is the concentration in the ion exchanger; e.g. millicuries per gram.

\( C_{\text{Sol}} \) is the concentration in the water at equilibrium; e.g. millicuries per milliliter.

From the definition of the distribution coefficient, the volume of water and the quantity of ion exchanger used, one can develop, for any storage facility, an expression which quantifies how the radioactive material sorbed by the ion exchanger will partition between the ion exchanger and the water; that is, what fraction, at equilibrium will be
on the ion exchanger. Relationships between the fraction on the exchanger, values of $K_D$, and quantity of ion exchanger in equilibrium with the water are shown in Figure 4. The family of curves is useful in selecting how much exchanger should be used. Obviously, it is more efficient to use a quantity in the steep part of the curve before the "knee".

Based on the laboratory study, several tests were made incorporating Zeolon-100 in the basin water filter system. The results are shown in Table 2. The correspondence between the calculated fractions removed and the measured quantities was excellent and reaffirmed that radiocesium behaved in a predictable ionic fashion.

Addition of Zeolon-100 (in two-kilogram additions to the filter makeup media) was adopted for normal pool filter use. It continues to maintain radiocesium concentrations in the 2.56 million liter pool at the $10^{-5} \text{ to } 10^{-4}$ microcures per milliliter range; thirty-fold less than radiocesium concentrations common before its use.

This process modification is well suited to the Morris spent fuel storage operation. However, for other application, side effects would have to be evaluated. For example, a basin water sample, subjected to very sensitive analysis, contained a few parts per million of silicon which is not seen in the demineralized makeup water. This silicon presumably comes from the zeolite and might not be acceptable in other applications of this quite specific cesium removal agent.

4. COMMENT

During the course of their lifetimes, many commercial facilities undertake studies pertinent to their own operation and interests. The results of such studies may also be useful to others concerned with the technology - researchers, regulators, and other facility operators.


### TABLE 1

HYDROGEN CONCENTRATIONS IN AIR VOID OF SPENT FUEL SHIPPING CASKS

<table>
<thead>
<tr>
<th>Burnup MWD/MTU</th>
<th>Burnup MWD/MTU</th>
<th>Cooling Time Days</th>
<th>Cask Skin °F</th>
<th>Fuel Residence Time-Hours</th>
<th>Hydrogen Vol-Percent</th>
</tr>
</thead>
<tbody>
<tr>
<td>5/21/76</td>
<td>28,958</td>
<td>1125</td>
<td>95</td>
<td>72</td>
<td>0.19</td>
</tr>
<tr>
<td>5/21/76</td>
<td>26,324</td>
<td>1125</td>
<td>88</td>
<td>68</td>
<td>0.38</td>
</tr>
<tr>
<td>7/21/76</td>
<td>31,790</td>
<td>147</td>
<td>120</td>
<td>20</td>
<td>0.58, 0.38</td>
</tr>
<tr>
<td>7/22/776</td>
<td>33,160</td>
<td>148</td>
<td>120</td>
<td>16.5</td>
<td>0.67, 0.69</td>
</tr>
</tbody>
</table>

### TABLE 2

ZEOLON BASIN SUMMARY

<table>
<thead>
<tr>
<th>Run</th>
<th>ZEOLON-100 Kg</th>
<th>Initial Inventory Ci</th>
<th>Fraction Removed</th>
<th>KD(Cs)(3) 10^6 ml/g</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.91(2)</td>
<td>8.6</td>
<td>0.30</td>
<td>1.1</td>
</tr>
<tr>
<td>2</td>
<td>1.36(2)</td>
<td>6.1</td>
<td>0.35</td>
<td>1.0</td>
</tr>
<tr>
<td>3</td>
<td>1.36(2)</td>
<td>3.8</td>
<td>0.36</td>
<td>1.0</td>
</tr>
<tr>
<td>4</td>
<td>1.36(2)</td>
<td>2.7</td>
<td>0.38</td>
<td>1.2</td>
</tr>
<tr>
<td>5</td>
<td>4.54</td>
<td>1.8</td>
<td>0.68</td>
<td>1.2</td>
</tr>
<tr>
<td>6</td>
<td>2.27</td>
<td>4.4</td>
<td>0.40</td>
<td>0.8</td>
</tr>
<tr>
<td>7</td>
<td>2.27</td>
<td>2.7</td>
<td>0.43</td>
<td>0.9</td>
</tr>
</tbody>
</table>

(1) Based on KD(Cs) = 1 x 10^6 ml/g
(2) Filter contained 17 Kg cation exchanger. Correction made.
(3) Calculated from basin water parameters and measured radiocesium removal.
FIGURE 1:
Comparison of Calculated and Measured Heat Generation Rates
for Spent Fuel Stored at Morris Operation
FIGURE 2:
Comparisons of Measured and Predicted Heat Generation Rates for PWR Fuel Assemblies (0.4 MTU)
FIGURE 3:
Relationship between Concentration of Radionuclides in Pool Water and Measured Radiation Dose Rates

\[
S.E. = 0.23 (^{60}\text{Co} + 0.25 ^{137}\text{Cs} + 0.66 ^{134}\text{Cs}) + 0.9
\]
FIGURE 4:
Relationship between Quantity of Ion Exchanger, Distribution Coefficient, and Fraction Sorbed
APPENDIX A. FACILITY DESCRIPTION

The Morris Operation (MO) spent fuel storage facility has a present capacity of about 700 metric tons (Te) which is equivalent to the annual fuel discharge from 20-25 large nuclear steam supply systems. The fuel is stored under water in an interconnected three compartment "pool" containing about 675,000 gallons of water. Two of the compartments have a total area of about 2,800 square feet and the water is 28.5 ft deep. The third compartment is 312 square ft in area and contains a 48.5 ft deep unloading "pit". The pool is constructed of strongly reinforced concrete two to four ft thick and is lined with a stainless steel sheathing. It is equipped with a leak detection system, monitors and other auxiliary equipment.

The water serves to provide shielding from the (penetrating) gamma radiation, and acts as a medium for removal of heat produced by the decay of residual radionuclides.

During storage of spent fuel, a very small amount of radioactivity transfers from the spent fuel assemblies to the basin water. Therefore, a water purification system, consisting of a versatile pre-coat filter system is supplied to remove both particulate materials - which affect visibility - and dissolved chemical and radiochemical materials which may respectively provide a potential for corrosion and low, but undesirable, radiation fields in the vicinity of the basin water. The filter system consists of screen covered perforated tubes (septa) installed in a pressure vessel. The septa serve as a foundation for applying a layer - or layers - of filter agents. These materials are applied remotely by pumping a water suspension of the desired filter media through the septa. Removal of the spent filter is also accomplished remotely by simply reversing the flow and routing the spent
filter media to storage. Area of the filter is 115 square feet and the basin water is pumped through the filter at a rate of 250 gallons per minute.

The heat generated by the residual radioactive fission products is removed by circulating the basin water at flow rates up to 750 gallons per minute through a finned stainless steel heat exchanger over which air at ambient temperature is forced by large electrically driven fans.

At the pumping rates used the basin water is "turned over" in 45 hours by the filter system, and 15 hours by the cooling system.

The spent fuel is stored in baskets which consist of sections of pipe - roughly the length of a fuel assembly - closed at the bottom and welded together longitudinally to form a square array. Openings near the bottom of the containers are provided to allow convective flow of water. Two types of baskets are used, one consisting of nine eight-inch diameter pipes for Boiling Water Reactor (BWR) fuel assemblies, the other of four twelve-inch diameter pipes for Pressurized Water Reactor (PWR) assemblies. The baskets are equipped with latches which engage in positions of a strong grid-like fixture resting on the bottom of the storage pool. These latches lock the four corners of the basket to the supporting grid to provide a system designed and tested to meet the established seismic criteria for the site.

Immediately adjacent to the fuel storage basin is an area used for preparing incoming casks for unloading. Here samples are taken, the cask is flushed and other operations carried out.

The fuel basin adjoins a separate, enclosed "cask receiving area" capable of accommodating casks moved over highway or railroad.
Auxiliary equipment for the basin storage system includes cranes, grapples, underwater closed circuit television, lifting yokes, ventilation system, etc. In addition, a 600,000 gallon low level waste storage vault is used to accumulate aqueous wastes. An evaporator and associated off-gas control equipment are used to safely reduce volume of the vault's contents on a periodic basis. No contaminated wastewater is discharged to the environs.
APPENDIX B. SELECTED STUDY TOPICS AT MORRIS OPERATION

B.1 SPENT FUEL STORAGE SYSTEM

1. Personnel exposure in storage basin activities including spent fuel receiving and shipping.

2. Alternate cask decontamination methods.

3. Mechanical performance of an alternate basin water filter.


5. Stored fuel thermal power estimates.

6. Basin water cooler decontamination and characterization of internally deposited contamination.

7. Effectiveness of natural cesium (nitrate) in removing radiocesium from basin water cooler internal surfaces.

8. Identification of "leakers" in stored fuel.

9. Storage basin temperature profiles.

10. Rates of radioisotope transfer from fuel to pool water as a function of basin water temperature.

11. Correlation of basin water radionuclide concentrations and surface exposure rates.
12. Use of a synthetic zeolite to remove radiocesium from basin water.


14. Final decontamination of metallic objects by electropolishing.

15. Spent fuel radiation levels measured in air and water.


17. Minimization of contaminated waste water to reduce periodic evaporation for volume reduction.


B.2 TRANSPORTATION SYSTEM

1. Hydrogen gas in IF-300 Casks.

2. Flush sample analyses after 20 days residence of spent fuel in an IF-300 Cask.

3. Radiation exposure levels from loaded IF-300 Casks.

4. Spatial distribution of cask contamination during the fuel unloading operation.

5. Electropolishing large objects - IF-300 Cask Redundant Yoke cradle.
6. Demonstration of helium leak testing the IF-300 Cask closure.

7. Repair of IF-300 damaged sealing surface.


B.3 FACILITY SYSTEMS

1. Utility steam boiler optimization.

2. Performance evaluation of water demineralizer system.

3. Ventilation system conversion from "once-through" to partial recirculation and resulting energy savings.

4. Use of small electric steam boiler to enable summer shutdown of the main utility boiler.

5. Using a heat pump to provide space heating from spent fuel decay heat.

B.4 OTHER

1. Radiation profile of BWR control rods.

2. Radiation measurements of high level reactor waste curtains.
STORAGE OF IRRADIATED ADVANCED GAS COOLED REACTOR FUEL IN
CEGB REACTOR PONDS

K.A. Simpson

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Berkeley Nuclear Laboratories,
Berkeley, Gloucestershire, GL13 9PB, England
All civil stations of the Advanced Gas Cooled Reactor (AGR) type in the United Kingdom discharge irradiated fuel to on-site pond storage. The fuel elements are irradiated in stacks of eight, called stringers. After discharge the end fittings of the stringer are removed in the irradiated fuel dismantling cell and the linking 'tie bar' cropped so that the elements can be separated and discharged to the cooling pond where they are stored for periods of ~ 100-200 days in unpainted skips made from 304 steel. The elements are held vertically in separate compartments by means of skip inserts made from boronated steel.

An AGR fuel element (Boocock et al, 1982) is approximately 1 m long and consists of 36 pins held in three rings by a grid and two braces, the pin cluster being located in a double walled graphite sleeve. The pin cladding and grid material is a 20%Cr/25%Ni stainless steel stabilized with niobium. The brace material is a titanium stabilized 20%Cr/25%Ni stainless steel. The fuel consists of annular pellets of high density uranium dioxide (10.8 g cm$^{-3}$) with a grain size of ~ 10 μm.

Typical irradiation conditions to date in the carbon dioxide coolant involve pin cladding temperatures ranging from ~ 350°C to 750°C up a fuel stringer and peak fuel temperatures of less than ~ 1200°C. The current mean discharge irradiation is 18 GWD/teU achieved over some five years.

Since the skips holding the elements during pond storage provide incomplete physical restraint of the fuel, criticality control in pond is achieved via the pond water. This consists of demineralized water dosed with boron in the form of boric acid to a level of 1250 g.m$^{-3}$ boron. The pH is then brought back to 7 with caustic soda (~ 100 g.m$^{-3}$).
Pond water treatment is provided by means of cooling, filtration and ion exchange (Windsor, 1982). The non-regenerable mixed bed ion exchange units use resins Amberlite IRN77 and IRN78 to remove activity, mainly cationic, and control aggressive anions, mainly chloride. Typical flow rates through the ion exchange plant give pond turnover times of a few days and resin lifetimes of several months. The pond make-up rate is equivalent to very roughly one pond volume per year, water being lost through evaporation and operations associated with pond water treatment plant routines and servicing.

No problems have been experienced with the operation of the current water treatment plant. The cooling leads to pond water temperatures of 25°C. Very little particulate material has arisen and water clarity is excellent. Chloride levels in pond water are controlled to < 0.5 g.m⁻³.

The radioactive content of the pond water is regularly monitored and an example of a typical monthly return for a pond receiving fuel at the current design channel average discharge irradiation of 18 GWD/teU is given in Table 1, rounded to one significant figure to illustrate the main features of the activity pattern.
TABLE 1

POND WATER ACTIVITY LEVELS (M Beq m\(^{-3}\))

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Filter Inlet</th>
<th>Filter Outlet</th>
<th>IX Unit Outlet</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sc-46</td>
<td>0.4</td>
<td>0.3</td>
<td>0.3</td>
</tr>
<tr>
<td>Cr-51</td>
<td>20</td>
<td>20</td>
<td>6</td>
</tr>
<tr>
<td>Mn-54</td>
<td>10</td>
<td>10</td>
<td>0.3</td>
</tr>
<tr>
<td>Co-58</td>
<td>5</td>
<td>5</td>
<td>N/D</td>
</tr>
<tr>
<td>Co-60</td>
<td>7</td>
<td>7</td>
<td>0.3</td>
</tr>
<tr>
<td>Zn-65</td>
<td>0.6</td>
<td>0.6</td>
<td>N/D</td>
</tr>
<tr>
<td>Sb-124</td>
<td>5</td>
<td>5</td>
<td>5</td>
</tr>
<tr>
<td>Cs-134</td>
<td>10</td>
<td>10</td>
<td>10</td>
</tr>
<tr>
<td>Cs-137</td>
<td>6</td>
<td>6</td>
<td>6</td>
</tr>
</tbody>
</table>

N/D - detectable.

Nuclides arising from irradiation of steel constituents were expected to be seen in pond as a consequence of leaching from clad oxide formed in-reactor and possibly from corrosion of steel components of the fuel elements in pond. The cation resin was selected to remove these and it performs satisfactorily. However experiments in which irradiated fuel pins were immersed in simulated pond water have indicated that Co-60 release from the pin cladding cannot account for the levels seen in pond water. The activity of sleeve graphite has been characterized in a detailed study which distinguishes activity arising from activation of originally inactive material deposited from the reactor coolant on the large graphite surface area. Measurements of Co-60 release into simulated pond water from immersed samples of irradiated sleeve graphite
with low and high amounts of deposit have shown that releases from graphite are greater than from fuel cladding and confirm the importance of the deposit in contributing to Co-60 release.

The presence of caesium in the AGR pond water was not anticipated, since it was planned to bottle any fuel detected as failed in reactor before discharge to the pond, and caesium is not removed to any extent by the ion exchange resins as shown in Table 1. Caesium selective resins suited to AGR pond water chemistry are available and could incorporated into the treatment scheme if required. The Table also shows a consistent observation that the ratio of the activities Cs-134: Cs-137 is > 1. The ratio of activities when the isotopes are produced by fission is a function of irradiation conditions but for AGR fuel it is less than unity. The study of sleeve graphite mentioned above showed that both Cs-134 and Cs-137 were present after irradiation in amounts which could be accounted for by the initial impurity levels of inactive caesium, and uranium. Leaching tests using irradiated graphite samples into simulated pond water confirmed the release of caesium in amounts which could explain both the level and activity ratio observed in pond water.

The release of activity from fuel exposed to pond water either by failure to bottle fuel defected in reactor or by clad breaching in pond has been studied using deliberately defected irradiated AGR pins immersed in simulated pond water at cave ambient temperature. Small defects e.g. a 1 mm diam hole in the cladding over the fuelled region of the pin, have been shown to release activity, predominantly caesium, very gradually over periods of hundreds of days. It has been confirmed that with defects of this type the central void in the pin can become water-logged and that the activity concentration in the internal solution is much greater than in external water, suggesting activity release is diffusion controlled and likely to be defect dependent. As expected release is faster from pins which have been decapped to give direct exposure of fuel cross section and central void to water, and slower from
pins in which the defect was made in an end cap such that the solid sintox pellet separated the defect from the fuel. The total caesium activity entering the water has been compared with the measured and predicted rare gas release or gap inventory. The long lived rare gas gap inventory is seen to be an upper limit on the fractional release of caesium to pond water for typical residence times in CEGB ponds.

There is no evidence of fuel pin failure in pond. The fuel elements can be inspected during storage by underwater colour photography and by use of an underwater colour television camera and are generally seen to be in excellent condition. Slight visual signs of corrosion are observed on elements where the steel components have been irradiated in a temperature range conducive to sensitization (Flowers, 1977). For example, parts of the brace which include welded strip sections are affected. A small number of elements are selected annually for detailed examination in cave following pond storage. There is no evidence of general corrosion in pond. Metallography of pins irradiated at temperatures conducive to sensitization has shown slight intergranular attack extending to only one or two grains' depth, ~ 20 µm. Intergranular attack has also been seen, on metallographic examination, associated with those areas of the brace where corrosion was visually apparent in pond, but this has not significantly impaired brace integrity.

Electrochemical techniques are being developed for in-pond use aimed at monitoring local changes in pond water chemistry in the vicinity of the fuel, and at characterizing fuel component behaviour. For example a metal probe/reference electrode assembly has been used to measure the rest potential of fuel element components in pond. The conductivity of the water is sufficient for the fuel element components and skip to behave as a coupled system displaying a uniform potential equal to that of the 304 steel skip.
Monitoring of fuel behaviour and activity release in storage ponds will be extended to AGR stations undergoing commissioning and under construction.

SUMMARY AND CONCLUSIONS

On-site fuel cooling ponds associated with the Advanced Gas Cooled Reactors in the UK have now achieved several years of operation without problems over maintenance of recommended pond water chemistry, water clarity, activity control or fuel storage behaviour. The likely sources of activity release into the ponds have been assessed. Fuel performance is monitored by in-pond inspection and limited destructive examination of elements after storage.

ACKNOWLEDGEMENTS

This paper summarizes the work of many people at these laboratories including J.A. Crofts, L. Felstead, D. Hatton, M. Phillips, D. Reed, I.H. Robins and S.P. Tyfield. The paper is published by permission of the Central Electricity Generating Board.
REFERENCES


CURRENT DESIGN OF WATER POOL TYPE ISFSI

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ABSTRACT

Current engineering/design of water pool type Independent Spent Fuel Storage Installations (ISFSI) show that this method of spent fuel storage can be economically competitive with dry storage concepts. Based on these results and its technical merits, water pool storage should be strongly considered for future storage capacity expansion. This conclusion is based on engineering studies, site specific design, and license preparation for a water pool ISFSI meeting the requirements of 10 CFR Part 72. A factor which contributes to water pool storage cost estimates being lower than those reported several years ago, is cost savings due to location of the ISFSI on an existing reactor site. Many existing site characterization data and environmental reports can be used for licensing. Also, existing organization and facilities can be utilized for such items as training, security, management, fire protection, water, sewer, etc. Another factor contributing to lower costs is the use of current U.S. design and regulatory standards for the storage of at least one-year-old spent fuel in an ISFSI as opposed to using nuclear reactor standards.

Proven water pool technology, safety, licensability, operations, and now a reference design, can result in a shortened lead time for a water pool type ISFSI. While water pools may not be the best choice for all spent fuel storage needs, current designs indicate that this concept provides flexibility, safety, and favorable economics.
The purpose of the work and operating experience which is being reported at this workshop is to provide adequate licensable spent fuel storage capacity until such time as reprocessing or geological repository facilities are available to accept this fuel. In our work as an engineering/design organization, we have addressed the problem of designing new storage facilities and modifying existing facilities for many clients both in the U.S. and overseas. Since each client usually has different requirements and restrictions, we have investigated and developed several concepts for new spent fuel storage including a concrete dry storage cask, dry vault, and water pool. Today I would like to discuss the result of our work in the development of a reference design for a water pool type Independent Spent Fuel Storage Installation (ISFSI). This design is easily modified to meet specific site conditions and interface requirements with any nuclear facilities. It served as the basis for a complete site specific wet pool ISFSI design, safety analysis, environmental review, and preparation of a complete license application meeting the requirements of 10CFR72 which is the U.S. regulation governing this type of facility. To our knowledge, this effort is the first completed in the U.S. for the design and preparation of a license application for a new, site specific wet pool spent fuel storage installation meeting current design and regulatory requirements. The results of this work have illustrated that independent spent fuel pools can provide the flexible and economic storage of decayed spent fuel.

2. G/C WATER POOL DESIGN FEATURES

The water pool ISFSI design consists of four major areas: cask receiving/handling, the spent fuel pool, support equipment, and personnel support and access control area. The cask receiving/handling area is
equipped with a 125 ton remote, radio controlled cask handling crane and auxiliary hoist. The layout of this area facilitates cask handling by providing space for storage and laydown of cask impact limiters, tie-downs, lifting yokes and inspection equipment. The cask receiving/handling area also contains provisions for cask washdown, preparation for cask unloading, and cask decontamination after removal from the pool. The layout arrangement prevents heavy objects, such as the cask, from passing over the spent fuel pool.

In response to operational experience gained at the G.E. Morris Operations and at West Valley, access platforms around the cask are included in the design, and automatic decontamination spray nozzles are located in both the decontamination area and the cask unloading pool. For situations where the facility has high cask receipt rate requirements, the cask preparation and decontamination can be designed to be performed by robotic equipment to decrease cask handling time and operational personnel radiation exposures.

The cask loading/unloading pool is directly adjacent to the cask decontamination area. To decrease the maximum possible cask drop height and the building height needed for cask clearance, a removable gate is provided between the cask loading/unloading pool and the decontamination area. The cask loading pool and its associated gates are designed so that cask lids and the gates do not have to be removed from the pool water during normal operations, thus minimizing the spread of radioactive contamination. The cask pool can be drained and filled by recycling water from the cask pool water storage tanks.

The spent fuel pool and the fuel handling equipment which serves the pool area are typical of what is found in many reactor spent fuel pools. The pool is approximately 40 ft deep with stationary high density storage racks. A fuel handling crane is used to remove fuel from the cask and to place the fuel into the storage racks. This type of design is utilized because many utilities desire to have similar operations at
both the reactor pool and the ISFSI. Alternatively, the pool depth could be reduced by approximately 14 ft and movable storage racks utilized. These racks would be loaded in a deep cask pool and then moved into storage in the shallower storage pool.

The spent fuel pool and the facility are designed to maintain a minimum of 8 ft of water above the stored fuel during all credible accident and design natural phenomenon conditions. Future expansion of storage capacity is available through modular additions to the facility. Interfaces between existing and future storage pools have been identified and the design facilitates such as expansion.

The spent fuel storage rack design utilized is a free-standing, high-density, non-poisoned design which takes credit for a minimum spent fuel burnup in the subcriticality analysis. Both the pool and the storage rack structures are designed for 2:1 consolidated fuel rods.

The support equipment area includes equipment for: pool water cooling, makeup and purification, low-level radioactive waste collection, utilities, fire protection, radiation monitoring, and HVAC. The design requirements for these systems are greatly reduced in comparison to their counterparts at a reactor spent fuel pool. For example, due to the five-year decay time for the spent fuel and the large thermal mass of the pool water, the spent fuel pool cooling system can be unavailable for long periods of time and still maintain conditions required for the safe storage of spent fuel. As such, the cooling system does not need an emergency power supply, nor is it designed for redundancy. Due to the quiescent nature of the spent fuel storage and facility operations, the only functional requirements needed during off-normal conditions are to maintain a subcritical configuration of spent fuel under a minimum shielding depth of water, and to monitor for radioactive releases. While some systems are designed to fail in a safe condition (such as fuel and cask cranes) it is not necessary that they function during off-normal conditions.
The personnel and access control facilities are located adjacent to the main building. The arrangement allows for ease of modification to accommodate various site and client requirements without major impact on the remainder of the facility layout. For the site-specific design, this area included a security office, a health physics office, an office for records and their retention, a locker room, a clean shop area, a sample room for radioactive analysis, a personnel decontamination area, a control room, and an emergency power battery room. The layout of this part of the installation was developed to allow for efficient personnel flow and control of access into the ISFSI. The control room was located in this area to facilitate daily operator surveillance without having to enter the controlled access areas of the main building as this facility is normally unmanned except for once a day operator surveillance.

3. ISFSI ESTIMATED COST

As part of G/C's design and licensing effort for a water pool ISFSI, a detailed estimate of the overall facility cost was prepared for a specific site application. The capital cost for the first module of storage was estimated at $28.2 million in constant 1984 dollars, with an average annual operating cost of approximately $0.58 million per year over the life of the facility. The storage capacity of the module was approximately 990 MTU of unconsolidated fuel or 1980 MTU of consolidated fuel rods. The capital cost estimate included the cost of sitework, labour and material for constructing the facility and its associated systems, equipment and ISFSI system interfaces with the on-site nuclear plant. The estimate also included engineering fees and allowances for temporary construction facilities, owner's management costs, construction management costs, and a 20 percent contingency.
The basis for the direct construction costs was the ISFSI layout drawings, system flow drawings, and system descriptions. Manufacturer's budgetary quotations were utilized in the pricing of major equipment. Prices for material and other equipment were obtained from catalogues, standard reference publications, and vendors. Construction labour costs were developed assuming union craft labour rates applicable to the area of the ISFSI site. Labour productivity reflect construction of a new facility separated from an operating nuclear power plant.

The cost of engineering was based on actual G/C engineering cost to design and perform all analyses for the ISFSI and prepare the license application, plus the proposed cost quoted to the client for final engineering.

The average annual operational cost was based on man-hour estimates for various operations such as fuel transfer, routine maintenance, operator surveillance, security, low-level radioactive waste processing, and special operation modes. In addition, the annual cost includes operation materials such as consumables, spare parts, and resin. Based on average power requirements for the installation equipment, and average makeup water demands, the annual utilities costs were determined. Finally the operational cost includes packaging, transportation, and burial of low-level radioactive wastes generated by the ISFSI. The unit values for these operation items were based on known client costs or were estimated based on typical utility costs. If rod consolidation is to be performed in this facility, the operational cost for the process would need to be added.

There are several reasons why we believe these estimates provide an accurate picture of what the actual cost can be for a water pool ISFSI located on the site of an existing nuclear facility. Unlike several studies, the G/C capital cost estimate is not based on an empirical formula with storage capacity as its only variable. In addition, the annual operational cost was not estimated as a simple percentage of
capital costs. The G/C cost estimates were based on an engineered
design. Also the intent of the cost development was to estimate the cost
of the ISFSI at the midpoint of the expected cost range. Some previous
estimates have been developed on a "not to exceed" cost basis. This type
of approach can result in a conservatively high cost estimate.

In addition to having a design basis for the G/C cost estimate,
the design itself offers both capital and operational cost savings when
compared to some previous pool storage concepts. Due to the decayed
nature of spent fuel stored in an ISFSI, the design requirements of
10CFR72 and ANS 57.7 are significantly less stringent than those for a
reactor storage pool. These differences in design requirements can be
translated directly into economic savings. Therefore, the estimated cost
of the G/C ISFSI, which is designed to 10CFR72 and ANS 57.7 requirements,
should be lower than cost estimates for facilities based on 10CFR50 and
ANS 57.2 design requirements.

The use of an existing nuclear plant site for the ISFSI site is
another cost savings in comparison to a location at a new site away from
the reactor. Many times existing site data can be used for design and
licensing without the expense of gathering site characterization data.
The license application preparation is easier and less expensive because
of existing approved environmental reports for the site. The cost of the
ISFSI is reduced by utilization of existing plant systems such as site
radiation monitoring, security, fire protection, water, sewer,
demineralized water, and other systems depending on the particular site.
Operational costs are also reduced by utilization of existing management,
training, health physics, and operation organizations.

Finally, we believe that operational input from existing
facilities and previous design experience have produced an ISFSI design
which tends to reduce the cost of the facility and allow ease of
operations and maintenance. Operation experience from both the
G.E. Morris Operations and West Valley were factored into the design. In
addition, G/C has designed spent fuel pool storage for more than 20 nuclear facilities including reactor spent fuel pools and five independent or adjacent spent fuel pool storage facilities. This experience has been used to produce a design and cost estimate which represents a credible effort on which to base future decisions regarding the use of pool storage for interim storage of spent fuel.

4. CONCLUSION

Current engineering and design work on a site specific water pool ISFSI has shown that this demonstrated technology can be a flexible, economic solution for the safe interim storage of spent fuel. The estimated cost of these facilities is significantly lower than that predicted during the later 1970's. This result is based on an engineered design, safety analysis, and site specific license application. Due to the desirable qualities of demonstrated technology, operation experience, flexibility, and favourable costs, the use of a water pool ISFSI should be reassessed in light of today's design improvements. We believe that water pool storage is as viable as many of the dry storage concepts and in some cases it may be the preferred method to provide additional spent fuel storage capacity.
STORAGE OF IRRADIATED FUEL IN BRITISH
NUCLEAR FUELS PLC (BNFL) PONDS*

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* Paper submitted too late for presentation at the Workshop
1. INTRODUCTION

British Nuclear Fuels plc (BNFL) provides a complete nuclear fuel service for reactor operators throughout the world. In addition to fuel manufacture and enrichment, BNFL has been reprocessing fuel at its Sellafield site for over 30 years. During that time, over 25,000 te of irradiated fuel has been reprocessed, representing fuel from Magnox stations within the UK and overseas. Additionally, some 1300 te of oxide fuel has been received for reprocessing.

2. AGR FUEL

a. BNFL's policy in respect to storage of irradiated fuel is to provide a short duration "buffer" capacity prior to reprocessing. Fuel will be pond stored for around 2/3 years from reactor discharge to facilitate cooling and the decay of short-lived fission products after which it will be dismantled in preparation for reprocessing.

b. AGR receipts from the UK Electricity Boards are now building up towards the THORP programme which is scheduled to be operational from 1990. Reactor discharges are currently around 18 GWD/te and are expected to increase progressively to 24 GWD/te stringer average.

c. Since 1977, a programme of routine Post Storage Examination (PSE) has been undertaken in which selected fuel assemblies have been dismantled and examined. This has been supported by underwater visual examination of elements in skips and by analyses of water chemistry. It is planned to continue this effort to characterize and monitor the condition of full burn-up fuel when this is available. To date, no in-pond pin cladding failures have been observed.
d. A 10 year storage period was originally estimated to be a prudent maximum for storing substantial quantities of AGR fuel in water (i.e. risk of stainless steel cladding deteriorating with subsequent fission product release). Experience from more recent development work suggests that this period could be extended by careful control of water chemistry.

e. Current developments include the design and construction of major new receipt and storage facilities, namely Pond 5, which is due to be commissioned in 1985. This plant will handle both Magnox and AGR fuel.

3. WATER REACTOR FUEL

A number of water reactor BWR/PWR elements have been examined, mostly by visual examination with some metallography. Fuel has been stored in ponds for upwards of 13 years with no evidence of in-pond degradation of fuel pins, confirming world experience in respect to zircaloy clad fuel.

4. MAGNOX FUEL

In view of Magnox corrosion, it was not intended to store Magnox fuel for long periods in water. A storage period of about 6 months before reprocessing is envisaged. To minimize corrosion effects, the pond water chemistry is being controlled.
SESSION 3

DRY STORAGE TECHNOLOGY AND ENGINEERING STUDIES

Chairman: D. Deacon
GEC - Energy Systems Limited
ABSTRACT

Over several decades essentially all types of irradiated nuclear fuel have been shipped and/or handled under dry conditions in hot cells. Essentially all types of irradiated fuels have been stored in dry facilities, some of which began operation as early as 1964. Four major dry storage concepts have developed: concrete silos (canisters), dry wells, metal casks, and vaults. Irradiated fuel has been emplaced in and retrieved from facilities representing all four concepts. Dry storage has been conducted safely and with relatively low radiation doses to the operators. Three of the four storage concepts have been licensed, indicating that dry storage is a relatively mature technology.

Currently, dry storage is expected to fill a complementary but important role to wet storage for interim storage of water reactor fuel in the US. Dry storage may fill expanded roles in future extended storage (e.g., monitored retrievable storage) and in repository disposal.

It is important that dry storage functions in a cooperative climate, where experience gained in one application is assimilated into other applications. It is also important to assess prospects to integrate current storage technologies with downstream fuel cycle activities to the extent possible.
1. INTRODUCTION

Dry storage is emerging as an attractive complement to wet storage for interim spent fuel management. While wet storage will almost certainly remain the predominant method for interim storage, unreprocessed fuel is likely to be stored dry in extended storage (e.g., MRS) or disposal scenarios. Dry storage has a substantial experience base, involving four major concepts; (Johnson, 1981; Johnson, 1983a); essentially all fuel types have been stored under dry conditions. Dry storage facilities already are licensed in several countries, indicating a relatively mature technology. This paper summarizes the dry storage experience base, outlines the rising interest in dry storage, and projects a perspective of eventual dry storage applications.

In addition to reviews of dry storage technology (Anderson, 1980; Johnson, 1983a; Ospina, 1983), technical developments have been reported in several conferences and publications, including the following:


2. DRY STORAGE EXPERIENCE - PAST AND PRESENT

2.1 HISTORICAL PERSPECTIVE

Figure 1 outlines significant phases in the history of dry handling and storage of irradiated fuel and indicates key dates in the development of dry storage technology. In addition to the more direct sources of dry storage data, e.g., demonstrations and licensed operations, fuel handling and examination in hot cells have provided insights to irradiated fuel behaviour in dry environments. There have also been opportunities to inspect and monitor the fuel during cask loading and unloading operations. Several fuel types, involving thousands of assemblies have been shipped under dry conditions. LWR fuel shipped by sea provides insights into fuel behaviour in the short-term in that the fuel resides in casks for times which sometimes exceed 100 days, with maximum cladding temperatures near 385°C (information supplied by K. Goldman, Transnuclear, Inc.).
2.2 DRY STORAGE CONCEPTS

Four major dry storage concepts have evolved (Anderson, 1980; Johnson, 1981; Johnson, 1983a).

- concrete silo (concrete canister)
- dry well (surface and deep types)
- metal cask
- vault

Concrete silos, dry wells, and metal casks are modular. Vaults sometimes involve major front-end construction, but conceptual designs of modular vaults also have been developed, e.g., Modrex (Doroszlai, 1982) and Modular Dry Storage (Baxter, 1982).

Cooling is passive for silos, dry wells and casks. Cooling for vaults may be natural or forced draft.

Figures 2-5 summarize world experience with the four dry storage concepts. Irradiated nuclear fuel was emplaced in most of the facilities listed. In a few facilities, electrical heaters were used to simulate nuclear heat for design verification. Figures 2 and 3 show electrically-heated tests for silos and dry wells. Metal cask development included numerous electrical heater tests (not shown on Figure 5). Electrical heater tests also were a factor to establish the licensing basis for the Wylfa vault.
In addition to numerous demonstrations and licensed dry storage operations with spent nuclear fuel, the dry storage data base includes laboratory and hot cell tests. Table 1 summarizes tests which contribute to the dry storage data base for Zircaloy-clad water reactor fuel.

Irradiated fuel has been emplaced and retrieved from facilities representing all four dry storage concepts. Facilities, representing three of the four storage concepts, have been licensed (Table 2). Almost all fuel types have been emplaced in dry storage facilities, including BWR,* PWR, PHWR, HTGR, Magnox, LMFBR, and several test reactor fuel types (AVR, DIORIT, EBR-II, ETR, JRR-3, WR-I). Fuel behavior in dry storage has been excellent. For example, cover gas monitoring on over 6500 irradiated LWR in inert gas tests and demonstrations has not shown evidence of even a single rod failure. (One LWR rod failed in an air test).

Tables 1 and 2 and Figures 2-5 indicate that dry storage already has a substantial history. However, currently only a minor fraction of the world spent fuel inventory is in licensed dry storage.

* See definition of terms in the Glossary.
3. EMERGING STATUS OF DRY STORAGE

3.1 STATUS OF FUEL MANAGEMENT OPTIONS

While reprocessing is underway in some countries, worldwide it has not developed on the scale or schedule originally expected. In 1982, about 20,000 MT of oxide fuel has been discharged and only a small fraction has been reprocessed (Rybalchenko, 1983). Therefore, most countries are expanding interim storage capacities. Table 3 summarizes the range of approaches to meet interim storage needs. Wet storage will continue to be the initial interim storage method at almost all nuclear reactors as long as the reactors operate. Many water-cooled power reactors have expanded wet storage capacities to accommodate between one and two decades of fuel discharges.

Current estimates suggest that by 2000 AD, more than 200,000 MT of oxide spent fuel will have been discharged (Rybalchenko, 1983). Delays in fast breeder reactor development suggest that reprocessing is likely to develop slowly over the next several years. Wet storage will continue to provide large fractions of utility interim storage requirements. However, numerous reactors indicate that they will exceed projected wet storage capacities between 1985 and 2000 AD.

Dry storage offers an attractive complement to wet storage. Dry storage capacity can be purchased on a modular basis, with relatively short procurement time, once licensing is in place. Most dry storage concepts are passively cooled. Currently, dry storage economics appear favorable, particularly for addition of relatively small incremental capacities (Johnson, 1981; Lawrence, 1982). Over the next decade dry storage will grow, but will remain subordinate to wet storage, probably through the end of the century. While subordinate to wet storage, dry storage will fill an important role for utilities which need to develop timely solutions to storage problems.
3.2 STATUS OF TECHNICAL CONSIDERATIONS

The fact that dry storage facilities have been licensed in several countries indicates that the technology is relatively advanced. However, several areas still under evaluation are indicated below.

3.2.1 Fuel Packaging

Some dry storage concepts require that fuel is canned, due to emplacement and retrieval requirements. However, uncanned fuel has been successfully emplaced and retrieved, involving the metal cask concept in the FRG. Canning also is not used for GCR fuel at Wylfa and for LMFBR fuel stored in Idaho dry wells. Failed water reactor fuel currently remains in wet storage. Eventually, decisions regarding how to package the fuel for extended storage and/or disposal will be necessary unless the fuel is reprocessed.

3.2.2 Storage System Monitoring

All dry storage facilities have monitoring strategies to verify that storage conditions are as prescribed. As dry storage emerges into an era of broader application, it is important to examine monitoring needs to assure that the facilities are neither over-monitored nor under-monitored. Fuel temperature monitoring is accomplished with thermocouples in cask demonstrations. The data are used to verify the accuracy of heat transfer codes. Validated heat transfer codes preclude the need for routine temperature monitoring. A monitoring procedure for inert cover gases has been demonstrated in the Federal Republic of Germany (Fleisch, 1982). The method involves monitoring pressure levels in a plenum between sealed lids which separate the fuel from the
surrounding environment. Monitoring of cover gas quality by intermittent extraction of cover gas samples has been conducted for dry storage demonstrations. The cover gas analyses enable detection of impurity species and levels in inert cover gases, and detection of fission gas release to determine whether cladding breaches develop during storage. Monitoring the cover gas quality may be unnecessary for LWR fuel when air is the cover gas or when temperatures decrease to a level where rates of fuel reactions with oxidizing species are not significant. Some vault facilities (e.g., INEL and Wylfa) monitor cover gases for humidity or for certain impurities (Anderson, 1980, Johnson, 1983a).

3.2.3 Impacts of Failed Fuel

Essentially all fuel types are subject to development of cladding breaches in a small fraction of the fuel rods (plates, elements, etc.) during reactor service. In some cases, the type and location of the breaches are known, but in some cases they are not. Uncanned water reactor fuel with cladding breaches can be stored in water without significant impacts. Metallic fuels (e.g., Magnox) with breaches are less tolerant of extended water storage.

Breached fuel can be stored in inert gases because there is no mechanism to propagate the cladding failure nor to degrade the exposed fuel. Satisfactory behavior in inert gas has been demonstrated for Zircaloy-clad BWR fuel with reactor-induced defects (Johnson, 1983b). Acceptable storage conditions for oxide fuel (e.g., AGR, CANDU, LWR) in air are being investigated.
3.2.4 Cover Gases

Cover gases for spent fuel in demonstration and operational facilities include air, argon, carbon dioxide, helium, and nitrogen. For magnesium-clad metallic uranium Magnox fuel, carbon dioxide is used for freshly-discharged fuel; older fuel can be stored in air with proper moisture control. Nitrogen was used for two FRG demonstrations (Kaspar 1982) and for some dry shipments. Most fuel types: water reactor, LMFBR, test reactor, are stored in inert gases. However, both air and nitrogen are being evaluated for storage of Zircaloy-clad water reactor fuel. Air also is being evaluated in the United Kingdom as a cover gas for AGR fuel (Simpson, 1983). Canadian fuel temperatures have been relatively low in dry storage demonstrations (<120°C) (Naqvi et al, 1982). This is a temperature regime where preliminary evidence suggests that air will not react with UO₂ at cladding defects at rates which will degrade fuel and cladding.

There is an interest among utilities which plan to store oxide fuel in inert gases, to define conditions where monitoring for inertness would no longer be required. If an acceptable fuel storage temperature for air can be defined, a significant reduction in monitoring effort could result.

Stainless steel-clad gas reactor and FBR fuel becomes severely sensitized during reactor exposure. There is evidence that the sensitized cladding will degrade by intergranular corrosion in oxygenated water or in moist air, even at relatively low temperatures. (Long, 1984). Stainless-clad LWR fuel operates during reactor service in a thermal regime where sensitization occurs very slowly. Therefore, stainless-clad LWR fuel is expected to be relatively resistant to degradation in water or moist air.
3.2.5 Fuel Rod Consolidation

Consolidation involves removal of rods from assembly hardware, and placing them into close-packed arrays to provide more efficient use of storage space. There currently is interest in consolidation of LWR and AGR fuel. Consolidation demonstrations have been conducted in the USA with unirradiated fuel assemblies. Also, four irradiated PWR assemblies were consolidated in an Oconee reactor storage pool (Bailey, 1984).

Consolidation can be performed by either wet or dry methods. Both wet and dry storage can benefit from space savings resulting from the close-packed arrays.

When evaluating the rod consolidation option for a specific application, consideration must be given to its impacts on other pool operations, rod accountability, methods for hardware disposal and the problem of handling of broken rods. Economics of consolidated storage remain to be clearly established. Currently, there is very little experience on which to base definitive economic assessments.

3.2.6 Economic Considerations

Cost estimates have been generated for several dry storage concepts, and have been compared to wet storage costs (Johnson, 1981; Lawrence, 1982). The relative costs for the various storage methods vary with facility size. For utilities which need relatively small, incremental storage capacity additions, modular dry storage concepts show cost advantages over wet storage, according to the published cost estimates. However, somewhat streamlined wet storage designs are emerging, with estimated costs which compare more favorably with dry storage (Johnson, 1984).
3.2.7 **Integration of Fuel Cycle Activities**

As alternative fuel management concepts develop, it is important to consider how they can be implemented to integrate efficiently with downstream fuel cycle activities. Impediments to integration center on uncertainties regarding the downstream scenarios. Will the fuel cycle be open? or closed? or both? Estimates suggest that by 2000 AD, the energy represented in North American spent fuel will exceed 80 y of oil consumption at current rates. Dwindling oil supplies seem likely to augment the resolve to recover and utilize the energy. Even with the current climate of uncertainty, it is important to analyze near-term fuel management solutions with respect to how they can best integrate with downstream actions.

4. **DRY STORAGE IN THE FUTURE**

The basis has been established for expanded use of dry storage to meet interim storage needs over the next few decades. In the USA, monitored retrievable storage (MRS) is an emerging concept. If the MRS is approved by the US Congress, the favored concepts involve dry storage (Hall, 1984).

Some countries (the FRG, Sweden, Finland, and the USA) are investigating spent fuel disposal in repositories. It currently is not clear what fraction of the spent fuel will be placed in repositories. Spent fuel disposal technology is in developmental stages. Sweden is investigating LWR fuel assembly encapsulation in copper (Ahlstrom, 1984). The FRG and USA are considering fuel encapsulation in metal canisters having inert cover gases (Closs, 1984; USDOE, 1989). Some aspects of that technology already have been investigated at the Climax granite mine on the Navada Test Site, where eleven PWR spent fuel
assemblies were emplaced in deep dry wells, approximately 425 m underground (Ballou, 1982). The assemblies have been removed from the Climax site and are undergoing further evaluation at the EMAD site.

It is important that dry storage functions in a cooperative climate, where experience gained in one application is assimilated into other applications. It is also important to assess prospects to integrate current storage technologies with downstream fuel cycle activities to the extent possible.

ACKNOWLEDGMENTS

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<tr>
<th>Acronym</th>
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<tr>
<td>AECL</td>
<td>Atomic Energy of Canada, Ltd.</td>
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<tr>
<td>AGR</td>
<td>advanced gas-cooled reactor</td>
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<td>ANL</td>
<td>Argonne National Laboratory, Argonne, Illinois</td>
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<td>AVR</td>
<td>Arbeitsgemeinschaft Versuchs Reactor, GmbH</td>
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<td>BCL</td>
<td>Battelle Columbus Laboratories, Columbus, Ohio</td>
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<td>BWR</td>
<td>boiling water reactor</td>
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<tr>
<td>CANDU</td>
<td>Canadian natural-uranium, heavy-water-moderated and -cooled power reactor</td>
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<tr>
<td>CSFM</td>
<td>Commercial Spent Fuel Management, USDOE Program conducted by PNL</td>
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<td>DIORIT</td>
<td>Swiss heavy water research reactor, Wuerenlingen, Switzerland</td>
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<td>DOE</td>
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<td>EBR-II</td>
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</tr>
<tr>
<td>EMAD</td>
<td>Engine Maintenance Assembly and Disassembly hot cell facility, NTS</td>
</tr>
<tr>
<td>EPRI</td>
<td>Electric Power Research Institute, Palo Alto, California</td>
</tr>
<tr>
<td>ETR</td>
<td>Engineering Test Reactor, INEL</td>
</tr>
<tr>
<td>FBR</td>
<td>fast breeder reactor</td>
</tr>
<tr>
<td>FRG</td>
<td>Federal Republic of Germany (West Germany)</td>
</tr>
<tr>
<td>GCR</td>
<td>gas-cooled reactor</td>
</tr>
<tr>
<td>GNS</td>
<td>Gesellschaft fur Nuklear-Service, FRG</td>
</tr>
<tr>
<td>HEDL</td>
<td>Hanford Engineering Development Laboratory, Richland, Washington</td>
</tr>
<tr>
<td>HTGR</td>
<td>high-temperature gas reactor</td>
</tr>
<tr>
<td>IAEA</td>
<td>International Atomic Energy Agency, Vienna, Austria</td>
</tr>
<tr>
<td>INEL</td>
<td>Idaho National Engineering Laboratory, Idaho Falls, Idaho</td>
</tr>
<tr>
<td>JRR-3</td>
<td>Japanese Research Reactor No. 3, Tokai, Japan</td>
</tr>
<tr>
<td>KWO</td>
<td>Kernkraftwerk Obrigheim, FRG</td>
</tr>
<tr>
<td>KWU</td>
<td>Kraftwerk Union, Erlangen, FRG</td>
</tr>
<tr>
<td>KWW</td>
<td>Kernkraftwerk Wuergassen, FRG</td>
</tr>
<tr>
<td>LMFBR</td>
<td>liquid metal fast breeder reactor</td>
</tr>
<tr>
<td>LWBR</td>
<td>Light Water Breeder Reactor (Zircaloy-clad Shippingport reactor core)</td>
</tr>
<tr>
<td>LWR</td>
<td>light-water reactor</td>
</tr>
<tr>
<td>Magnox</td>
<td>magnesium alloy cladding for gas reactor fuel; typical composition is Mg-0.8 wt% Al-0.0025 wt% Be</td>
</tr>
<tr>
<td>MDS</td>
<td>modular dry storage</td>
</tr>
<tr>
<td>MRS</td>
<td>monitored retrievable storage</td>
</tr>
<tr>
<td>NTS</td>
<td>Nevada Test Site, USA</td>
</tr>
<tr>
<td>OH</td>
<td>Ontario Hydro, Toronto, Ontario, Canada</td>
</tr>
<tr>
<td>PHWR</td>
<td>pressurized heavy-water reactor</td>
</tr>
<tr>
<td>PNL</td>
<td>Pacific Northwest Laboratory, Richland, Washington</td>
</tr>
<tr>
<td>PWR</td>
<td>pressurized water reactor</td>
</tr>
<tr>
<td>REA</td>
<td>Ridihalgh, Eggers and Associates, Columbus, Ohio</td>
</tr>
<tr>
<td>TN</td>
<td>Transnuclear (or Transnuklear)</td>
</tr>
<tr>
<td>TVA</td>
<td>Tennessee Valley Authority</td>
</tr>
<tr>
<td>UK</td>
<td>United Kingdom</td>
</tr>
<tr>
<td>USA</td>
<td>United States of America</td>
</tr>
<tr>
<td>WNRE</td>
<td>Whiteshell Nuclear Research Establishment, Pinawa, Manitoba, Canada</td>
</tr>
<tr>
<td>WR-1</td>
<td>Whiteshell Reactor 1; heterogeneous, uranium, organic-cooled</td>
</tr>
<tr>
<td>Wy1fa</td>
<td>Magnox GCR at Wy1fa, Wales, United Kingdom</td>
</tr>
<tr>
<td>Zr</td>
<td>Zirconium</td>
</tr>
<tr>
<td>Zry-2</td>
<td>Zircaloy-2</td>
</tr>
<tr>
<td>Zry-4</td>
<td>Zircaloy-4</td>
</tr>
<tr>
<td>Country</td>
<td>Organization</td>
</tr>
<tr>
<td>---------------</td>
<td>--------------</td>
</tr>
<tr>
<td>Canada</td>
<td>AECL/OH</td>
</tr>
<tr>
<td>Canada</td>
<td>AECL/OH</td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td>FRG (a)</td>
<td>KWU</td>
</tr>
<tr>
<td>FRG (a)</td>
<td>KWU</td>
</tr>
<tr>
<td>FRG/Italy (a)</td>
<td>Ispra/NUKEM</td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td>FRG (a)</td>
<td>KWU</td>
</tr>
<tr>
<td>USA</td>
<td>BCL/HEDL</td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td>USA</td>
<td>BCL/HEDL</td>
</tr>
<tr>
<td>USA</td>
<td>BCL/PNL</td>
</tr>
<tr>
<td>USA</td>
<td>PNL</td>
</tr>
<tr>
<td>USA</td>
<td>EPRI/TVA/ HEDL/ANL</td>
</tr>
<tr>
<td>USA</td>
<td>NRC/HEDL/ EG&amp;G</td>
</tr>
</tbody>
</table>

(a) Unpublished data supplied by J. Fleisch, DWK, Hannover, FRG.
(b) Isothermal for 190 days; then four rods were transferred to 570°C.
(c) Unirradiated rods; however, under test conditions, radiation effects would have been fully annealed.
### TABLE 2:

**LICENSED DRY STORAGE FACILITIES**

<table>
<thead>
<tr>
<th>Country</th>
<th>Site/Facility</th>
<th>Status</th>
<th>Fuel Type(s)</th>
<th>Cladding</th>
<th>Burnup MWd/MTM</th>
<th>Initial Maximum Cladding Temperature °C</th>
<th>Storage System</th>
</tr>
</thead>
<tbody>
<tr>
<td>Canada</td>
<td>Whiteshell Nuclear Res. Est.</td>
<td>First storage, 1975</td>
<td>Organic-cooled test reactor, WR-1</td>
<td>1r-2.5 MO</td>
<td>6,000</td>
<td>120 (est.)</td>
<td>Concrete Canister (silo)</td>
</tr>
<tr>
<td>Federal Republic of Germany:</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>· Wuergassen</td>
<td></td>
<td>Two-year demo, completed</td>
<td>BWR</td>
<td>1r-2</td>
<td>78,000</td>
<td>385</td>
<td>Metal Cask</td>
</tr>
<tr>
<td>· Karlsruhe</td>
<td></td>
<td>Demo started March 1983</td>
<td>PWR</td>
<td>1r-4</td>
<td>39,000</td>
<td>430</td>
<td>Metal Cask</td>
</tr>
<tr>
<td>· Julich</td>
<td></td>
<td>Demo started 1984</td>
<td>PWR</td>
<td>1r-4</td>
<td>NA</td>
<td>330</td>
<td>Metal Cask</td>
</tr>
<tr>
<td>· Garleben</td>
<td></td>
<td>First storage expected in 1984</td>
<td>PWR/WR</td>
<td>1r-4/2 1r-4</td>
<td>Range</td>
<td>Range</td>
<td>Up to 420 metal casks in concrete building; capacity, 1500 MTMM</td>
</tr>
<tr>
<td>· Ahaus</td>
<td></td>
<td>Construction License; 1984</td>
<td>--</td>
<td>--</td>
<td>Range</td>
<td>Range</td>
<td>Metal Casks; capacity, 1500 MT</td>
</tr>
<tr>
<td>Switzerland</td>
<td>Wuerenlingen</td>
<td>First storage, June 1983</td>
<td>OIGNF/TT Test reactor</td>
<td>1r-4</td>
<td>17,000</td>
<td>180 (est.)</td>
<td>Metal Cask</td>
</tr>
<tr>
<td>United Kingdom</td>
<td>Wylo</td>
<td>First storage, 1971</td>
<td>Magnox Mg-alloy</td>
<td></td>
<td>5,000</td>
<td>150</td>
<td>Vault</td>
</tr>
</tbody>
</table>

NA - Not Available.
TABLE 3:
CURRENT STATUS OF PLANS TO MEET STORAGE NEEDS

A. Interim Storage Needs to be Met by Adding Wet Storage Capacity: (examples)
   - Bulgaria
   - Czechoslovakia
   - Finland
   - Sweden

B. Major Use of Dry Storage to Complement Wet Storage
   - Federal Republic of Germany (Gorleben and Ahaus sites, 1500 MT each)

C. Secondary Use of Dry Storage to Complement Wet Storage
   - USA (Licensing actions initiated)
   - UK (Wylfa)

D. Dry Storage Used to Solve Specific Storage Needs
   - Canada - WR-1
   - FRG - GCR (AVR)
   - France - LMFBR
   - Japan - Test Reactor
   - Switzerland - Test Reactor
   - USA - LMFBR, GCR, Radioactive Scrap

E. Investigating Dry Storage and Disposal
   - Argentina - study for Atucha
   - Canada - central storage facility design study
   - FRG - fuel disposal
   - Italy - study for Trino
   - Spain - developing dry cask technology
   - Sweden and Finland - fuel disposal
   - USA - Monitored Retrievable Storage (MRS) and fuel disposal
   - USSR - purchased and loaded CASTOR V
   - UK - central storage facility design study
FIGURE 1:

History of Dry Handling and Storage of Irradiated Fuel
<table>
<thead>
<tr>
<th>LOCATION</th>
<th>FUEL TYPE</th>
<th>OPERATIONAL HISTORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>NEVADA TEST SITE - EMAD</td>
<td>PWR</td>
<td>64 65 66 67 68 69 70 71 72 73 74 75 76 77 78 79 80 81 82 83 84</td>
</tr>
<tr>
<td>WASHINGTON-HANFORD</td>
<td>ELECTRIC</td>
<td></td>
</tr>
<tr>
<td>CANADA-WNRE</td>
<td>WR-1</td>
<td></td>
</tr>
<tr>
<td>(13 CONCRETE CANISTERS</td>
<td>PHWR</td>
<td></td>
</tr>
<tr>
<td>IN PLACE - 1984)</td>
<td>ELECTRIC</td>
<td></td>
</tr>
</tbody>
</table>

**FIGURE 2:**

Dry Storage Experience in Concrete Silos; Adapted from Wright, 1981
<table>
<thead>
<tr>
<th>LOCATION</th>
<th>FUEL TYPE</th>
<th>OPERATIONAL HISTORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>NEVADA TEST SITE-</td>
<td></td>
<td>64 65 66 67 68 69 70</td>
</tr>
<tr>
<td>CLIMAX (GRANITE)</td>
<td>PWR</td>
<td>71 72 73 74 75 76</td>
</tr>
<tr>
<td>EMAD (SOIL)</td>
<td>PWR</td>
<td>77 78 79 80 81</td>
</tr>
<tr>
<td>EMAD (SOIL)</td>
<td>ELECTRIC</td>
<td>82 83 84</td>
</tr>
<tr>
<td>IDAHO-INEL (SOIL)</td>
<td>LMFBR</td>
<td></td>
</tr>
<tr>
<td></td>
<td>HTGR</td>
<td></td>
</tr>
<tr>
<td></td>
<td>LMFBR</td>
<td></td>
</tr>
<tr>
<td></td>
<td>RADIOACTIVE MATERIALS</td>
<td></td>
</tr>
<tr>
<td></td>
<td>LWBR (SCHEDULED FOR LATE 84)</td>
<td></td>
</tr>
<tr>
<td>KANSAS-PSV (SALT)</td>
<td>ETR/ELECTRIC</td>
<td></td>
</tr>
<tr>
<td>BASALT SOIL</td>
<td>ELECTRIC</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>WASHINGTON-HANFORD BASALT</td>
<td>ELECTRIC</td>
<td></td>
</tr>
<tr>
<td>SOIL</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

FIGURE 3:

Dry Storage Experience in Dry Wells, Adapted from Wright, 1981
<table>
<thead>
<tr>
<th>LOCATION</th>
<th>CASK TYPE</th>
<th>FUEL TYPE</th>
<th>OPERATIONAL HISTORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>FRG-WUERGASSEN</td>
<td>CASTOR Ic</td>
<td>BWR</td>
<td></td>
</tr>
<tr>
<td>FRG-JULICH</td>
<td>TN</td>
<td>GCR (AVR)</td>
<td></td>
</tr>
<tr>
<td>FRG-JULICH</td>
<td>CASTOR Ia</td>
<td>PWR</td>
<td></td>
</tr>
<tr>
<td>FRG-KARLSRUHE</td>
<td>CASTOR Ib</td>
<td>PWR</td>
<td></td>
</tr>
<tr>
<td>FRG-BIBLIS</td>
<td>TN-1300</td>
<td>PWR</td>
<td></td>
</tr>
<tr>
<td>FRANCE</td>
<td>CASTOR SPX</td>
<td>FBR</td>
<td></td>
</tr>
<tr>
<td>USSR LENINGRAD</td>
<td>CASTOR V</td>
<td>PWR</td>
<td></td>
</tr>
<tr>
<td>USA - GE MORRIS</td>
<td>REA 2023</td>
<td>BWR</td>
<td></td>
</tr>
<tr>
<td>SWITZERLAND-WUERENLINGEN</td>
<td>CASTOR Ic</td>
<td>TEST REACTOR</td>
<td></td>
</tr>
</tbody>
</table>

**FIGURE 4:**

Dry Storage Experience in Metal Casks
<table>
<thead>
<tr>
<th>LOCATION</th>
<th>FUEL TYPE</th>
<th>OPERATIONAL HISTORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>NEVADA TEST SITE-EMAD</td>
<td>PWR</td>
<td>64 65 66 67 68 69 70</td>
</tr>
<tr>
<td>WYLFA, WALES</td>
<td>MAGNOX</td>
<td>71 72 73 74 75 76 77</td>
</tr>
<tr>
<td>IDAHO-INEL</td>
<td>LMFBR</td>
<td>78 79 80 81 82 83 84</td>
</tr>
<tr>
<td>FRANCE</td>
<td>VITRIFIED WASTE (AVM)</td>
<td></td>
</tr>
<tr>
<td>TOKAI, JAPAN</td>
<td>TEST REACTOR</td>
<td></td>
</tr>
<tr>
<td>FUKUSHIMA, JAPAN</td>
<td>BWR</td>
<td></td>
</tr>
</tbody>
</table>

**FIGURE 5:**
Dry Storage Experience in Vaults, Adapted from Wright, 1981
ONSITE STORAGE OF SPENT NUCLEAR FUEL
IN METALIC SPENT FUEL STORAGE CASKS

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H.S. McKay

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Richmond, Virginia
USA 23219

ABSTRACT

Virginia Electric and Power Company (Vepco) owns and operates two nuclear power stations within its system: the North Anna Power Station located in Louisa County, Virginia; and the Surry Power Station located in Surry County, Virginia. Each of these power stations has two pressurized water reactor operating units which share a common spent fuel pool. Under the Nuclear Waste Policy Act of 1982, Vepco is responsible for providing interim spent fuel storage until availability of the Federal Repository. Vepco has studied a number of options and has developed a program to provide the required onsite interim spent fuel storage.

Options considered by Vepco included reracking, pin consolidation, dry storage and construction of a new spent fuel pool to provide the increased spent fuel storage capacity required. Vepco has selected reracking at North Anna combined with dry storage in metal spent fuel storage casks at Surry to provide the required onsite spent fuel storage. A dry cask storage facility design and license application were developed and the license application was submitted to the NRC in October, 1982. The selection of the option to use dry cask storage of spent fuel at Surry represents the first attempt to license dry storage of spent nuclear fuel in the United States. This storage option is expected to provide an effective option for utilities without adequate storage space in their existing spent fuel pools.
1. INTRODUCTION

Virginia Electric and Power Company (Vepco) owns and operates two nuclear power stations within its system: the North Anna Power Station located in Louisa County, Virginia; and the Surry Power Station located in Surry County, Virginia. Each of these power stations has two pressurized water reactor operating units which share a common spent fuel pool. The nuclear units at North Anna and Surry are all Westinghouse designed three loop PWRs; the units at Surry use 15 x 15 fuel and the units at North Anna use 17 x 17 fuel.

Under the Nuclear Waste Policy Act of 1982, Vepco is responsible for providing interim spent fuel storage until availability of the Federal Repository. All of the spent nuclear fuel which has been generated at this two stations is still stored in the fuel pool at these units. However, as indicated in Figures 1 and 2, fuel storage requirements will exceed the available capacity in the spent fuel pools at both stations prior to the projected availability of the Federal Repository in 1998. Vepco has studied a number of options for providing interim spent fuel storage until availability of a Federal Repository or Monitored Retrievable Storage facility, and has developed a program to provide the required onsite interim spent fuel storage.

2. INTERIM SPENT FUEL STORAGE OPTIONS CONSIDERED

Vepco has considered several options for providing increased capacity for spent fuel storage at the Surry and North Anna Power Stations. These options included: reracking the spent fuel pools with neutron absorbing spent fuel storage racks, fuel pin consolidation, transshipment of spent fuel between Surry and North Anna, dry spent fuel
storage, and construction of new spent fuel pools at Surry or North Anna. These options were evaluated on the basis of costs, technical feasibility, schedule for implementation and flexibility to respond to changes in storage requirements.

The spent fuel pools at the Surry and North Anna Power Stations were originally designed assuming that spent fuel would be shipped offsite within a few months of discharge for reprocessing. Thus, the pools only provided capacity for one full core and one reload batch from each reactor (total of 2 and 2/3 cores). When it became obvious that reprocessing would be delayed, both the North Anna and Surry spent fuel pools were reracked with high density fuel racks. The storage capacity with these racks is indicated in Figures 1 and 2 as the current capacity of the spent fuel pools. (Note that in Figures 1 and 2 one full core can still be discharged when the pool reaches the indicated capacity.) Both pools were also evaluated for reracking with neutron absorbing spent fuel racks. Based on this evaluation, the decision was made to rerack North Anna for a second time and an application to rerack North Anna with neutron absorbing spent fuel racks was filed with the NRC in August, 1982. These neutron absorbing spent fuel racks will increase the capacity of the North Anna spent fuel pool as indicated in Figure 2 with the line marked "reracked spent fuel pool capacity."

These new fuel racks are designed to store both North Anna and Surry spent fuel and will provide storage capacity to allow transshipment of up to 500 spent fuel assemblies from Surry to North Anna. The new fuel racks at North Anna are also designed to allow storage of consolidated spent fuel. If consolidation is developed as an effective and economical storage alternative, the new fuel racks, combined with consolidation, can provide for interim storage of North Anna spent fuel until well after the projected availability of the Federal Repository or Monitored Retrievable Storage. However, the analysis of the Surry spent fuel pool indicated that another reracking of this pool or storage of
consolidated spent fuel could not be performed because of structural limitations of the spent fuel pool. Therefore, the analysis of options for Surry centered on transshipment, dry storage or a new spent fuel pool.

As indicated above, the fuel racks at North Anna can be used to store Surry spent fuel and provide interim spent fuel storage for Surry by utilizing transshipment until 1990. In July, 1982, Vepco filed an application with the NRC to allow receipt and storage of up to 500 Surry spent fuel assemblies at North Anna. However, Louisa County and a citizens group intervened against both this application and the application to rerack North Anna. Louisa County has also enacted an ordinance prohibiting storage of radioactive materials in Louisa County unless the wastes were generated in Louisa County.

Vepco challenged Louisa County's ordinance and this ordinance was declared unconstitutional on March 3, 1983, by the United States District Court for the Eastern District of Virginia. However, Louisa County appealed this decision. Vepco has recently reached an agreement with Louisa County to limit transshipment of spent fuel by providing storage at Surry for Surry spent fuel as soon as a facility can be licensed and constructed at Surry. Louisa has withdrawn both its appeal of the court decision on its ordinance as well as its intervention in licensing proceedings for reracking and storage of Surry spent fuel at North Anna. An ASLB proceeding is continuing on the reracking and transshipment with the Louisa County citizens group.

Vepco has analyzed the options of both dry storage and construction of a new spent fuel pool to provide the increased spent fuel storage capacity required for Surry. Based on both cost and schedule, the decision to pursue dry storage was reached. The dry storage option selected by Vepco was dry metal spent fuel storage casks. A dry cask storage facility design and license application were developed and the
license application was submitted to the NRC in October, 1982. Dry casks provide a modular storage technique that, once licensed, can be implemented as needed.

Vepco has also been awarded a Co-operative Agreement program with the DOE to assist in the development of licensed dry cask storage. The Co-operative Agreement was signed on March 29, 1984 and provides for development and licensed dry cask storage at Surry supported by research and development activities at a Federal site using the same cask designs as are proposed for Surry. This project is also being supported by EPRI.

3. DESCRIPTION OF THE DESIGN OF VEPCO'S PROPOSED DRY CASK STORAGE FACILITY

An artist's conception of the design of Vepco's dry cask storage facility is shown in Figure 3. As shown in this drawing, the facility will consist of concrete pads on which the dry storage casks will be stored vertically. The concrete pads will be approximately 10 by 70 meters and will be approximately 1 meter thick. Each pad will be designed to store 28 casks and the pads will be constructed as needed to store fuel from Surry. There will be no active safety related systems on the storage pads; however, monitoring will be provided for cask internal or interlid pressure with a local alarm to indicate if any leakage of gas from the casks is occurring.

Each cask will hold approximately 10 metric tons of spent nuclear fuel and an average of 3 storage casks will be required each year to accept the spent fuel being discharged from the two Surry reactors. The spent fuel loaded into the storage casks will be spent fuel which has been discharged at least five years before transfer to dry storage.
Sufficient spent fuel will be transferred to the dry storage facility to maintain adequate space in the spent fuel pool to accept a full core discharge at any time.

Approximately 40 storage casks will be required to provide sufficient interim spent fuel storage until the projected availability of either Monitored Retrievable Storage or the Permanent Repository.

4. **FUEL TO BE STORED**

The fuel proposed to be stored in the facility will be Westinghouse designed 15 x 15 fuel which, as indicated above, will have decayed in the fuel pool for at least five years before transfer to dry storage. This will allow the fuel decay heat to drop to no more than one kW per fuel assembly. The limit on fuel enrichment will be 3.7 percent and the limit on burnup will be 35,000 MWd/MTU. The fuel will be stored in a helium atmosphere and the design of the casks will maintain the maximum fuel clad temperature for long term storage below 380 degrees centigrade except for short term situations, such as vacuum drying, when the temperature will be maintained below 450 degrees centigrade.

5. **DISCUSSION OF THE CASK DESIGNS SELECTED FOR TESTING**

Three cask designs have been selected for testing in the joint Vepco/DOE/EPRI program discussed above. The program plan calls for a total of up to four storage casks to be tested at a Federal site, and Vepco has now selected one cask from each of three vendors for testing. All of these are metallic storage casks; however, the designs and materials of construction differ for each of the casks. Casks are being
fabricated for the testing by GNSI, Woolley (REA), and Westinghouse. These casks include a cast nodular iron, cast steel, and stainless steel and lead cask.

The GNSI CASTOR-V cask is constructed from cast nodular iron and is designed to hold 21 PWR fuel assemblies. The cask uses a solid neutron shield which is placed into holes drilled axially into the cast iron wall. The cask iron provides the gamma shield and structural strength. Two stainless steel lids are used with a double metalic seal to retain the helium gas in the cask. The helium pressure in the space between lids is monitored to determine if there is any leakage from the cask.

The Westinghouse cask is constructed using a carbon steel shell and a stainless steel inner and outer lid. A metalic seal and a welded outer cover provide a double seal for the helium atmosphere in the cask. The neutron shielding is provided by use of a neutron shielding material, Bisco, outside the main carbon steel shell. The outer surface of the cask is a thin stainless steel shell which contains the Bisco material and provides a corrosion resistant cask exterior. The cask is designed for storage of 24 assemblies with a ten year decay time.

The Woolley (REA 2023) cask uses a lead gamma shield contained between two stainless steel shells with a water filled neutron shield tank around the outside of the cask. This cask is designed for storage of 24 fuel assemblies with a five year decay time.

Options for each of these three cask designs are being considered for storage of consolidated fuel. This offers the potential to double the storage capacity of each cask.
6. PROGRAM FROM PRECHARACTERIZATION OF FUEL

The fuel to be placed in to dry storage for Federal site testing has been precharacterized by examination of the operating records for the fuel cycles in which the spent fuel resided in the core, and by performing ORIGEN calculations to determine the decay heat being generated in the spent fuel. The spent fuel to be used for the testing program was also examined using a ultrasonic testing technique that determines fuel rod failure by checking for water in the fuel rods. This technique has been shown to be effective for fuel which has been discharged from the reactor for several years before testing. The fuel to be used in the initial dry storage tests will be fuel which has been confirmed not to have cladding defects through this testing technique.

7. REVIEW OF TESTING PROGRAM PLANNED FOR THE FEDERAL SITE

The testing program planned for these casks at the Federal site will involve transporting spent fuel from Surry to the Federal site in NRC licensed transportation casks and transferring this spent fuel to the storage casks discussed above. The initial tests for these casks will be performed using intact fuel. Spent fuel will then be consolidated at the Federal site for testing of dry cask storage of consolidated spent fuel. At least two casks will be used for testing of consolidated storage. This will require consolidation at the Federal site of 96 fuel assemblies to test two casks. Based on the results of these tests, Vepco plans to file an amendment to the dry cask storage facility license for Surry to allow storage of consolidated spent fuel at Surry. Consolidation in dry storage casks will be economical if the fuel from the two storage casks can be consolidated for less than the cost of the second storage cask.
8. TRANSPORTATION OF SPENT FUEL

Vepco is also interested in the potential future use of the storage casks being acquired for Surry for transportation of spent fuel to the repository or Monitored Retrievable Storage facility. While this transportation will be the responsibility of the DOE under the Nuclear Waste Policy Act, the large storage casks offer a potentially attractive option for transportation to the repository if they can be licensed under 10 CFR 71. This of course would require the co-operation of the DOE and an economic incentive for the utility.

9. CONCLUSION

Vepco is pursuing several options to meet its obligations under the Nuclear Waste Policy Act of 1982 to provide interim storage of spent fuel. The selection of the option to use dry cask storage of spent fuel at Surry represents the first attempt to license dry storage of spent nuclear fuel in the United States. This storage option is expected to provide an effective option for utilities without adequate storage space in their existing spent fuel pools.
Figure 1

Projected Spent Fuel Assemblies
To Be Discharged From
Surry Power Station

NUMBER OF SPENT FUEL ASSEMBLIES

3500
3000
2500
2000
1500
1000
500


CURRENT SPENT FUEL POOL CAPACITY

887

YEAR
Figure 2

Projected Spent Fuel Assemblies
To Be Discharged From
North Anna Power Station

- POTENTIAL CAPACITY WITH CONSOLIDATION
- RERACKED SPENT FUEL POOL CAPACITY
- CURRENT SPENT FUEL POOL CAPACITY
SURRY POWER STATION
DRY CASK INDEPENDENT SPENT FUEL STORAGE INSTALLATION
DECAY HEAT AND HEAT TRANSFER PREDICTIONS
FOR SPENT FUEL STORAGE SYSTEMS

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ABSTRACT

Decay heat generation rates of PWR and BWR spent fuel assemblies predicted with the ORIGEN2 computer program are presented and compared to values measured with calorimeters. Predictions are in good agreement with available experimental data.

Temperatures of single assembly and multi-assembly spent fuel storage systems with both unconsolidated and consolidated fuel are predicted with the COBRA and HYDRA thermal hydraulic computer programs. Predictions are compared to limited experimental data.
1. INTRODUCTION

The Nuclear Waste Policy Act (NWPA) of 1982 assigns the U.S. Department of Energy (DOE) the responsibility for conducting 1) at-reactor demonstration programs in cooperation with the private sector, 2) a research and development (R&D) program at federal sites to collect data necessary to assist utilities in their licensing efforts, and 3) generic research and development of spent fuel storage techniques. The Commercial Spent Fuel Management Program at the Pacific Northwest Laboratory (PNL) operated by Battelle is assisting the DOE in meeting their responsibilities. Battelle is also assisting the Electric Power Research Institute (EPRI) in spent fuel storage R&D areas identified by EPRI as being important to the utilities they represent.

A critical parameter in the research, development, demonstration, and implementation of spent fuel dry storage systems is heat transfer performance. Of major importance is the need for experimental heat transfer data from tests and demonstrations to support the first few utility license applications for dry storage systems. It is also important that adequate modeling techniques be established during initial tests and demonstrations to minimize the need for further expensive, time consuming experimental efforts. Because the decay heat generation rate of spent fuel stored in dry systems is a primary input parameter to heat transfer analyses, decay heat prediction methods should also be evaluated.

Once adequate data become available to support early licensing efforts and evaluate heat transfer modeling techniques, satisfactory methods should be documented and submitted to the Nuclear Regulatory Commission (NRC) for approval/qualification for use in performing dry storage licensing analyses. This paper summarizes R&D activities sponsored by DOE and EPRI at PNL/Battelle in pursuit of this goal. Modeli
efforts in the areas of decay heat and heat transfer in unconsolidated and consolidated spent fuel are presented.

2. DECAY HEAT

Accurate decay heat predictions or measurements are critical to spent fuel storage system heat transfer analyses since decay heat generation is a primary input to heat transfer predictions. Two studies have been completed which compare ORIGEN2 decay heat predictions to experimental spent fuel calorimeter data. The first was a PWR study sponsored by the DOE and conducted by the Hanford Engineering Development Laboratory under Contract to PNL (Schmittroth, 1984). The second was a BWR study sponsored by EPRI and performed by Battelle and General Electric Morris Operations. Results of each study are presented and discussed in this section.

2.1 PWR DECAY HEAT PREDICTIONS

Since 1980, five Turkey Point Reactor PWR spent fuel assemblies have been calorimetered at the DOE owned Engine Maintenance, Assembly, and Disassembly (EMAD) facility operated by Westinghouse and located on the Nevada test site. ORIGEN2 predictions of the decay heat generation rates of the fuel assemblies were completed and reported by Schmittroth et al., 1980. A water boil off calorimeter described by Creer and Shupe, 1981 was used to obtain the experimental decay heat measurements.

Table 1 presents calorimeter data, ORIGEN2 predictions, and comparisons of the two sets of information. As indicated, ORIGEN2 predictions compare to the calorimeter data within ±6% with the exception of fuel assembly D-04 and the second measurement on D-15. It is known that thermal equilibrium had not been established in the calorimeter for the D-04 measurement, thus making the prediction appear high (Schmittroth
et al., 1980). There is reason to believe the same is true for the high
D-15 prediction, since this prediction falls between two predictions
where agreement is good.

In 1981, eight PWR fuel assemblies from the San Onofre reactor
and six assemblies from the WEPCO reactor were calorimetered at GE-Morris
using an in pool calorimeter (Judson et al., 1982). Decay heat pre-
dictions using ORIGEN2 were also performed by HEDL and reported by
Schmittroth, 1984. Comparisons of predictions with data are presented in
Tables 2 and 3. The ORIGEN2 predictions agreed well (<7% difference)
with the experimental calorimeter data.

Predictions of decay heat rates for the San Onofre PWR assem-
blies were all higher than the calorimeter measurements, while all pre-
dictions for WEPCO assemblies were lower than data. No specific reason
for this result has been formulated, but it is interesting to note that
San Onofre fuel rods were stainless steel clad and the WEPCO assemblies
had Zircaloy cladding. Approximately 15% of the total decay heat of the
San Onofre assemblies was due to the relatively high content of cobalt 60
in the stainless steel clad, while less than 5% of the decay heat was due
to the lower content of cobalt in the Zircaloy-clad WEPCO assemblies.

It can be concluded that the agreement between the predicted and
measured decay heat values for the 19 PWR spent fuel assemblies is good,
and the separate consistencies for the San Onofre, WEPCO, and Turkey
Point results are remarkable. Except for the uncertainty caused by the
cobalt in San Onofre assemblies, the results indicate that a
calculational accuracy of ±5% or better can be achieved when reliable
input data are available.

Limitations of the existing data base should be noted, specifi-
cally the restricted range of cooling times studies (from 3 to 8 years).
Decay heat in this time period is dominated by fission products. For
long cooling times during which the actinide decay heat becomes
important, the present conclusions do not necessarily apply. In fact, there is reason to believe that substantial difficulties may exist in accurately predicting the actinide decay heat contribution.

2.2 BWR DEATH HEAT PREDICTIONS

During 1984, EPRI sponsored a decay heat study in which Battelle performed ORIGEN2 decay heat predictions of 11 BWR spent fuel assemblies and GE-Morris conducted calorimetry on the assemblies. Calorimetry was performed on one Dresden assembly and ten Cooper assemblies using a modification of the in-pool calorimeter used for the WEPCO and San Onofre PWR measurements (Judson et al., 1982). Results of the precalorimetry ORIGEN2 predictions are presented in Table 4 along with experimental data.

Comparisons of predicted to measured values indicate excellent agreement with the exception of assemblies D212, CZ102, and CZ259. The experimental difficulties associated with measuring the relatively low decay heat rates of D212 and CZ102 may have contributed to the relatively large difference between predicted and measured values. The difference between predicted and measured values for CZ259 was traced to a relatively high uncertainty (~18%) in the burnup for that assembly.

It appears that calculational accuracies of 5% or better are possible when reliable BWR input data are available. As in the case of PWR assemblies, it should be noted that the present BWR calorimetry data base is restricted to cooling times of 2 to 7 years, and the conclusions drawn from comparisons of predicted with measured values may not apply for significantly longer cooling times.
3. HEAT TRANSFER

Heat transfer predictions of existing sets of spent fuel storage system data and data yet-to-be-obtained have been completed or are in progress. These data sets include both single and multiassembly data, and pretest predictions consider both unconsolidated and consolidated spent fuel assemblies. The following sections briefly describe the heat transfer computer codes used, and discuss the predictions and comparisons between predictions and available experimental data for a few selected cases.

3.1 COMPUTER CODES

Up until a few years ago, most cask heat transfer analyses were performed with codes such as HEATING5 (Turner et al., 1977) or TRUMP (Edwards, 1972) in conjunction with the empirical correlation of Wooten/Epstein (Wooten and Epstein, Undated). HEATING5 and TRUMP are generalized three dimensional heat transfer computer codes utilizing the energy equation. Convection and radiation heat transfer are essentially considered as boundary conditions. Both steady state and transient heat transfer analyses are possible.

The codes are used to predict temperature profiles from a known boundary to the cask basket or fuel tube containing a spent fuel assembly. The Wooten/Epstein correlation is then used to predict peak clad temperatures of the fuel rods using code predictions for fuel tube surface temperatures. In most cases, only two dimensional analyses (R, θ) are performed since the analyses were for horizontal shipping casks, not vertical storage casks.

In 1982, a search for codes with the capability to accurately predict temperatures in a multiassembly cask using air in a vertical orientation was conducted under DOE sponsorship. That search led to the
decision to evaluate the COBRA and HYDRA codes for spent fuel storage heat transfer analyses. Versions of the codes could consider all three modes of heat transfer (conduction, convection, and radiation) because they included continuity, momentum, and energy equations. Pin-to-pin radiation heat transfer can be significant in dry storage systems, and both codes had versions that were capable of detailed radiation modeling. The main difference between the two codes is that COBRA is a finite volume code while HYDRA uses the finite difference approach. Therefore, HYDRA can be used to analyze heat transfer systems in relatively fine detail while COBRA uses a coarser nodding approach.

The codes are being used to predict operating conditions (temperatures) in storage systems containing both single and multiassembly unconsolidated and consolidated spent fuel. Vacuum, air, nitrogen, helium, and argon environments are being considered. Horizontal, vertical, and angles inbetween are being analyzed.

Because the codes can consider free convection, they can be used to locate instrumentation in a vertical orientation with convecting media such as air, nitrogen, or argon. It is very important that temperature sensors be placed near the axial locations where maximums will occur. Also, because the number of temperature sensors are limited, it is important that each sensor is placed in a position where a significant amount of information will result.

Each of the following sections present comparisons of predictions of HYDRA or COBRA, or both, with experimental data.

3.2 UNCONSOLIDATED SINGLE PWR SPENT FUEL ASSEMBLY

Comparisons of COBRA and HYDRA predictions with heat transfer data from an unconsolidated single PWR spent fuel assembly have been performed. The test section used by Westinghouse at the Engine Maintenance, Assembly, and Disassembly (EMAD) facility located on the Nevada Test Site
is described in detail by Unterzuber, 1981. Figures 1, 2, and 3 show details of the test section and PWR spent fuel assembly.

The test section (Figure 1) consisted of a PWR spent fuel assembly contained in an instrumented stainless steel canister held in place by an internal support cage. The canister and support cage were positioned in a liner which had band heaters attached to the outer surface. The band heaters permitted operation at selected peak clad temperatures.

The PWR spent fuel assembly (Figure 2) was a standard Westinghouse 15 x 15 rod array from the Turkey Point Reactor. The assembly contained 204 spent fuel rods, 20 control rod guide tubes, and an instrument tube. To measure temperatures inside the assembly, thermocouple trains inside thin tubes were inserted into guide tubes and the instrument tube. The shaded rods indicated in Figure 2 contained thermocouple trains at axial locations shown in Figure 3.

Tests at ~1.2 kW were run with the test section in a vertical orientation. Vacuum, helium, and air atmospheres were used. Radiation heat transfer is the primary mode of heat transfer in a vacuum environment while conduction is important with helium and convection is significant in air.

Figures 4 and 5 present comparisons of COBRA and HYDRA predictions with test data for the TC train located in the center instrument tube. The predictions presented were completed before the persons performing the analyses were aware of the experimental data. COBRA predictions (Figure 4) for vacuum and air were 10°C to 15°C lower than the data. COBRA over predicted the helium data by ~10°C. COBRA predicted the temperature profiles reasonably well for all these environments, especially the helium profile.
HYDRA predictions of the data (Figure 5) indicate over estimates (~<10°C) for vacuum, and under estimates (<10°C) for air and helium. Temperature profiles for the three backfills are good, with the vacuum and helium being exceptional.

The results of these analyses lead to the following conclusions:

- "Pre-look" predictions with COBRA and HYDRA compared well with experimental data.

- The continuity, momentum, and energy equations contained in the COBRA and HYDRA codes appear to adequately model conduction, convection, and radiation heat transfer that existed in the single spent fuel assembly.

- Radiation was the main mode of heat transfer in a vacuum environment, convection in an air backfill, and conduction in a helium backfill.

- Fuel rod temperatures were very sensitive to rod and canister wall radiation emissivities.

- Velocities with an air backfill were roughly an order of magnitude greater than helium velocities.

3.3 CONSOLIDATED SINGLE PWR SPENT FUEL ASSEMBLY

During 1984, EPRI sponsored a study to investigate the characteristics of heat transfer in consolidated PWR spent fuel (Cuta, Rector, and Creer, 1984). The purpose of the study was to predict effects of important parameters such as geometry and rod-to-rod, rod-to-wall, and wall-to-wall contact on peak clad temperatures in air/nitrogen and helium
environments. An unconsolidated PWR spent fuel assembly analysis was also conducted to serve as a reference to compare consolidated fuel results.

A cross section of the single assembly storage system modeled with the COBRA code is shown in Figure 6. An unconsolidated or a consolidated PWR assembly was assumed to be contained in a fuel tube. The fuel tube was positioned in a boundary wall having a uniform temperature. The assembly or fuel rods were those of a standard 15 x 15 PWR rod array. The unconsolidated assembly was assumed to be generating 1 kW and the consolidated assembly 2 kW. An axial view of the COBRA geometry model is shown in Figure 7.

It was assumed the consolidated rods and fuel tube could become nonaxisymmetric as shown in Figure 8. In addition to investigating rod-to-rod contact in the consolidated assembly, rod-to-wall, and wall-to-wall contact could be studied.

With air as the storage medium, the radial profile of the temperature difference in the consolidated assembly is not significantly different from that of the unconsolidated assembly. Figure 9 compares the radial profile of the temperature difference (from the peak rod to the boundary) for the unconsolidated assembly in air with that for the consolidated assembly in air with no contact conductance. Both assemblies have a relatively flat temperature profile across the rod array. In the consolidated assembly this is due mainly to the close packing of the rods. In the unconsolidated assembly the effects of radiation tend to smooth the profile to a nearly uniform temperature. The unconsolidated assembly has a less abrupt large temperature drop than the consolidated assembly between the edge rods and the fuel tube, since it does not have the thermal barrier of the canister wall with which to contend.

Figure 9 also shows that the overall temperature drop in air is significantly higher in the consolidated assembly than in the
unconsolidated assembly with the assumed geometry. This result is not unexpected, since the consolidated assembly has twice the power of the unconsolidated assembly, but the magnitude of the difference between the two cases indicates that heat transfer is somewhat less efficient in the consolidated assembly than in the unconsolidated assembly with air as the storage medium. This is because in the consolidated geometry, radiation and natural convection (the most efficient modes of heat transfer for air) are severely curtailed, leaving conduction as the primary mode of heat transfer.

Figure 10 compares the radial profile of the temperature drop for the consolidated and unconsolidated assemblies with helium as the storage medium. The total temperature drop (from the peak clad to the boundary) for the consolidated assembly is considerably smaller than in the unconsolidated assembly, even though the consolidated assembly generated twice the power. This result is in striking contrast to that obtained with air as the storage medium (see Figure 9) and illustrates the importance of the thermal characteristics of the storage medium for a consolidated assembly. Helium can transfer heat more efficiently by conduction in the consolidated geometry than by radiation and natural convection in the unconsolidated geometry, as the temperature drop is smaller for the consolidated case, despite generating twice the power. The different modes of heat transfer that dominate in the two cases result in significantly different shapes for the radial profiles shown in Figure 10. The temperature drop across the rods (from the center to a rod at the edge of the bundle) is less than 5°C in the consolidated assembly (about 6% of the total ΔT), compared to approximately 60°C (about 55% of the total ΔT) in the unconsolidated assembly. Even though the overall ΔT is lower in the consolidated assembly, most of the temperature drop occurs between the edge rods and the boundary. As a result, the two profiles actually cross at a point near the edge of the assembly. Beyond the edge of the assembly, where radiation and convection are dominant in both geometries, the profiles are nearly the same shape.
In the consolidated assembly, conduction is the dominant mode of heat transfer. Since helium is a much better heat conductor than air, it is a more efficient heat transfer medium than air in the consolidated geometry. Figure 11 illustrates this very clearly with curves comparing the effect of storage medium on the radial temperature drop in the consolidated assembly for the axisymmetric geometry (assuming no contact conductance). The two profiles have essentially the same shape, but the temperature drop in helium is only about a third of the temperature drop in air for the same power generation rate (2 kW). These results indicate that in the consolidated assembly with the assumed axisymmetric geometry, helium yields significantly more efficient heat transfer than air, primarily because of a higher thermal conductivity.

Since the consolidated assembly has twice the number of rods as the unconsolidated assembly, it is reasonable to assume that it may generate twice the amount of power. But for direct comparison with the unconsolidated geometry, it would be useful to know how the consolidated assembly behaves at the same power level. Comparisons of the temperature drop (from the peak rod to the boundary) in the consolidated and unconsolidated assemblies at 1 kW are shown in Figure 12 with air as the storage medium and in Figure 13 with helium.

Figure 12 shows that with air as the storage medium, the consolidated and unconsolidated bundles have nearly the same temperature drop from the peak rod to the boundary when they are at the same power. The consolidated assembly has a slightly higher overall $\Delta T$, since in the consolidated assembly the geometry severely restricts natural convection. This comparison indicates that radiation is the dominant mode of heat transfer when air is the storage medium, with some contribution due to natural convection, which is greatly influenced by the geometry.

By contrast, with helium as the storage medium, the enhanced conductance within the consolidated assembly results in a significantly lower temperature drop from the peak rod to the boundary. At the lower
power, the effect is quite dramatic, as Figure 13 shows. These results indicate that with the geometric assumptions used in this analysis, helium is the more efficient storage medium for both the unconsolidated and the consolidated geometry.

One should, however, resist jumping to the conclusion that consolidated fuel assemblies will be a superior mode of storage over unconsolidated assemblies. The smaller temperature drops with helium in the consolidated assembly do not imply lower peak temperatures in an actual multiassembly system with consolidated fuel generating more heat than unconsolidated assemblies. The results presented here are based on comparing the behavior of single consolidated and unconsolidated assemblies with identical boundary conditions. In actual practice, individual consolidated and unconsolidated assemblies generating different amounts of heat will not generally see the same canister or fuel tube temperatures, even in identical multiassembly storage systems. The temperature distribution in a storage system containing consolidated fuel will probably be quite different from that for a system containing unconsolidated fuel. A storage system with a number of consolidated canisters generating twice the heat of unconsolidated assemblies may have higher internal temperatures, regardless of how efficient the heat transfer might be in the individual canisters within the storage system. The temperature drop from the peak fuel tube to the environment will significantly influence the magnitude of the peak clad temperature. With twice as much heat to dissipate, a multi-assembly consolidated storage system will have up to twice the temperature drop from the fuel tube to the environment as the same system with unconsolidated fuel generating half as much heat. A complete evaluation of the consolidated spent fuel storage concept requires an analysis of the thermal performance of each system with consolidated and unconsolidated assemblies.

All of the predictions presented previously in Figures 9 through 13 assumed no rod-to-rod contact conductance. Effects of rod-to-rod contact conductance are shown in Figure 14, where radial temperature
differences are plotted for different values of contact conductance. The temperature profiles are taken at the axial location of peak rod temperature, from the hot rod to the boundary wall. The curves in Figure 14 are for air and a similar plot for helium is not included, since the very small variation with contact conductance yielded a singularly uninteresting profile. In every case approximately 80% of the total temperature drop in the assembly occurs across the region from the edge row of rods to the boundary. The air gaps between the rods and canister, and between the canister and fuel tube, plus the canister wall itself (with its assumed low emissivity of 0.2 for radiation) are the most significant thermal barriers to radial heat transfer in the assembly.

The assumption of axisymmetric geometry, while being conservatively hot, is quite unrealistic. Since there is nothing to prevent them from doing so, it is reasonable to expect the rods to touch the sides of the canister, and the canister to touch the fuel tube. The results for the axisymmetric geometry, which show a large fraction (50% to 80%) of the temperature drop occurring across the postulated gaps between rods and canister, canister and fuel tube, and fuel tube and boundary wall, lead one to expect enhanced heat transfer if the rods touch the canister, and if the canister touches the fuel tube. Calculations were performed with two different eccentric geometries to characterize the magnitude of the effect of geometry assumptions on predicted temperature differences.

Results of calculations with eccentric rods in an axisymmetric canister are shown in Figures 15 and 16 for both air and helium. Contact conductance was neglected in the cases shown. When neglecting the aid to heat transfer due to contact conductance, the eccentric geometry showed a smaller temperature drop than the axisymmetric geometry. In each case, the overall temperature drop decreases by approximately 6% with the change in geometric configuration. Eliminating the gap between the edge of the rods and the canister eliminates the sharp temperature drop from the edge rod to the canister, in both air and helium. With air, the
differences in slopes in this region between the two cases (see Figure 15) are so great that the curves actually cross each other.

The temperature difference within the closed canister, from the peak clad to the peak canister wall, decreases somewhat with increased contact conductance, as can be seen from the results in Table 5. The effect is more noticeable with air as the storage medium. These results show that the geometry assumed for rods within the canister has more impact on predicted temperature differences than assumptions concerning contact conductance. If the geometry is modeled correctly, the conservative assumption of no contact conductance can be expected to yield reasonable results for temperature differences of interest.

The results of calculations with eccentric rods in an eccentric canister are presented in Figures 17 and 18. Profiles of the radial temperature difference from the hot rod to the boundary wall, along a diagonal line from the hot rod which passes through the peak fuel tube wall temperature are shown. These are compared to temperature drop profiles for the axisymmetric geometry in air and helium. The curves for the fully eccentric geometry show, as expected, smaller overall temperature differences than do those for the axisymmetric geometry. In addition, they show a different shape, without the sharp temperature drop between the edge rods and the canister and between the canister and fuel tube. The eccentric geometry very effectively smooths the discontinuities calculated in the axisymmetric geometry.

The contact conductance has a greater effect in the fully eccentric geometry than in the axisymmetric geometry, since contact occurs not just rod-to-rod, but rod-to-wall and wall-to-wall. As Table 6 shows, the overall temperature drop decreases by 15°C between the case with zero contact conductance (209°C) and the case with reasonable contact conductance between all surfaces in contact (194°C) in air, and by 6°C for the same cases in helium. For the axisymmetric geometries, the same variation in rod-to-rod conductance alone produced changes in temperature drop
by only 8°C in air and 1°C in helium (see Figure 14). These results show that if it is assumed that the rods and walls touch in consolidated assemblies, the value selected for contact conductance can have an effect on the predicted temperature drop. This geometry provides the most efficient heat transfer path from the hot rod to the fuel tube possible in a consolidated assembly, and has the smallest temperature difference obtained in this study.

This analysis compares the performance of consolidated and unconsolidated spent fuel only in single-assembly geometries, and it is obvious that further analyses comparing the two storage methods in a cask or other multiassembly systems are required to fully assess their relative merits. Experimental investigations of consolidated spent fuel multiassembly storage systems are needed to provide data against which to evaluate computer code predictions. However, this analysis provides the following conclusions regarding consolidated versus un consolidated spent fuel storage:

- With air (low conductivity gases) as the storage medium, a consolidated fuel assembly may have a larger temperature difference between the hot rod and the fuel tube than an unconsolidated assembly, even if it has the same power. (If the consolidated assembly generates twice the heat, a much larger ΔT than in the unconsolidated assembly can be expected.)

- With helium (high conductivity gases) as the storage medium, a consolidated assembly will have a smaller temperature difference from the hot rod to the fuel tube wall than an unconsolidated assembly, even if it is generating twice as much heat. (At the same power, it will have a lower temperature difference.)

- The assumed geometric arrangement of the rods within a consolidated canister has a significant effect on the predicted temperature drop from the peak clad to the peak fuel tube
temperature. In this analysis, the overall ΔT decreased by more than 40% between cases with the most conservative and the most optimistic geometric assumptions, for both air and helium.

- If the geometry is assumed axisymmetric, with no rod-to-wall contact, rod-to-rod contact conductance has only a minor effect on the predicted temperature drop from the peak clad to the fuel tube.

- If the geometry is assumed eccentric with rod-to-rod, rod-to-wall, and wall-to-wall contact, the contact conductance can have an important effect on the overall temperature drop.

- Experimental data from single consolidated electrically heated assemblies and actual fuel assemblies are required to evaluate heat transfer computer codes for application to design and safety analysis. These data will provide evaluations of:
  
  - geometry models
  - contact conductance models
  - radiation and natural convection heat transfer models.

- Thermal-hydraulic analyses, with suitable computer codes, of dry storage systems containing consolidated fuel assemblies are required to accurately predict peak clad temperatures that may be encountered.

3.4 CASTOR-1C BWR STORAGE/TRANSPORT CASK

The thermal analysis of the unconsolidated single PWR assembly (section 3.2) was extended to the CASTOR-1C multiassembly BWR storage/transport cask. This DOE sponsored effort was aided by the cooperation
of DWK, Preussen Elektra, and GNS of the Federal Republic of Germany in supplying cask and fuel design details required to adequately model the cask in all three dimensions.

During 1982, 1983 and early 1984, DWK sponsored a test using a GNS CASTOR-1C BWR cask. The test was performed at Preussen Elektra's Wuergassen BWR reactor. A diagram of the cask is shown in Figures 19 and 20 (Fleisch and Ramcke, 1983).

The cask was loaded with 16 BWR spent fuel assemblies, eight 7 x 7 rod and eight 8 x 8 rod assemblies, generating ~1 kW each. The cask body and fuel assemblies were instrumented with thermocouples at axial locations indicated in Figure 19.

Comparisons of COBRA predictions of hot assembly temperatures and experimental data are shown in Figures 21 and 22. The axial temperature profiles at two decay heat rates shown in Figure 21 indicate excellent agreement between COBRA "pre-look" predictions and CASTOR-1C data. In the upper two-thirds of the hot assembly, predicted temperatures agree with data within 10°C. In the lower one-third of the assembly, predicted temperatures were as much as 40°C higher than experimental data which may indicate that predicted conduction heat transfer from the fuel assemblies to the cask bottom was not quite as high as the actual heat transfer.

Radial temperature profiles at the axial location of peak clad temperature (2.8 m) are shown in Figure 22. At the higher decay heat generation rate, predictions were ~10°C higher than data, and in general excellent agreement between predictions and data exists. At the lower decay heat generation rate, predictions essentially coincide with the experimental data.

Because predictions agree well with both the low and high decay heat generation rate data (low and high temperature data), it appears that COBRA permitted adequate modeling of all three modes of heat
transfer. Good agreement at the two heat generation rates is a very strong indication that radiation and conduction heat transfer were modeled well since they are the primary modes of heat transfer with a helium backfill.

3.5 REA BWR STORAGE CASK WITH CONSOLIDATED FUEL

The single consolidated PWR fuel assembly analysis conducted under sponsorship of EPRI (section 3.3) and the CASTOR-1C analysis with unconsolidated BWR fuel sponsored by DOE (section 3.4) were extended to an analysis of a REA cask (Figure 23) containing consolidated BWR spent fuel. This DOE sponsored analysis is thought to be the first attempt at a detailed analysis of a multiassembly spent fuel storage system containing consolidated fuel.

A cross section of the lead-stainless steel clad REA cask containing a liquid neutron shield is shown in Figure 24. The cask is approximately 8 feet in diameter and 16 feet long and can store 52 BWR spent fuel assemblies generating 0.4 kW each. The present analysis assumed consolidated fuel generating 0.8 kW could be loaded in the cask.

Figure 25 presents the axial temperature profiles in the hot assembly with unconsolidated and consolidated spent fuel in both nitrogen and helium. Consolidated fuel rods generating 0.8 kW per assembly in nitrogen were predicted to reach temperatures well over 300°C which is significantly higher than temperatures predicted in helium or for unconsolidated fuel. The profiles in Figure 25 indicate a larger temperature difference between unconsolidated and consolidated fuel in nitrogen than in helium. The larger temperature difference between consolidated and unconsolidated fuel in nitrogen indicates that free convection is severely restricted in close packed consolidated fuel rods as would be anticipated. The smaller temperature difference between consolidated and
unconsolidated fuel in helium indicates that free convection is not a primary mode of heat transfer in helium, and conduction in consolidated fuel is important.

Radial temperature profiles from the center of the cask to the ambient are presented in Figure 26. The profiles indicate the same differences between unconsolidated and consolidated spent fuel in nitrogen and helium as those of Figure 25. The profiles within an assembly show that consolidated fuel rods in helium are at a relatively uniform temperature in the radial direction as discussed in section 3.3. Flat radial profiles substantiate that a relatively high effective conductance exists in consolidated fuel rods in a helium backfill. It appears that storage systems with consolidated spent fuel will operate at significantly lower temperatures if gas backfills with relatively high thermal conductivities are used. In contrast, a comparison of the temperature profiles for unconsolidated spent fuel in nitrogen and helium indicate that if good convecting gases are used in relatively open basket designs, peak clad temperatures may not be significantly higher in gases with relatively high thermal conductivities but poor convecting properties.

The following conclusions resulted from this analysis:

- Consolidated fuel generating twice as much heat as unconsolidated fuel will most likely run hotter in multiassembly storage systems backfilled with low or high conductivity gases.

- Temperature differences between consolidated and unconsolidated fuel will be less in high conductivity backfill gases than low conductivity gases.

- The center-to-edge rod temperature difference is less in high conductivity backfill gas than a low conductivity gas.
• Convection is a secondary mode of heat transfer in a consolidated fuel assembly.

• Good experimental data are needed for licensing dry storage of consolidated fuel and evaluating prediction methods.

4. FUTURE ACTIVITIES

At the present time, four sets of experimental data have been identified to evaluate heat transfer codes such as COBRA and HYDRA. These data are 1) single unconsolidated PWR assembly, 2) single unconsolidated electrically heated PWR assembly, 3) CASTOR-IC cask with unconsolidated BWR fuel, and 4) REA BWR cask with unconsolidated fuel. Once "pre-look" predictions are completed in 1985, the computer analysts will be given the experimental data and "post-look" analyses performed if required. Reports for the single assembly, CASTOR-IC cask, and REA cask will be issued in 1985. Assuming the codes perform as expected, they will be documented and released during 1986. They will be submitted to the U.S. Nuclear Regulatory Commission in 1986. They will be submitted to the U.S. Nuclear Regulatory Commission in 1986 for approval to be used in performing licensing analyses.

ACKNOWLEDGMENTS

The support of the DOE, the PNL Commercial Spent Fuel Management Program, and EPRI is greatly appreciated by the author. Thanks are extended to DWK, Preussen Elektra, and GNS of the Federal Republic of Germany for providing details of the CASTOR-IC cask and BWR spent fuel. The outstanding work of the following individuals who actually performed the analyses reported herein is acknowledged.

ORIGEN2 - F.A. Schmittroth COBRA - C.L. heeler HYDRA - R.A. McCann
C.M. Heeb D.R. Rector L.E. Wiles
U.P. Jenquin N.J. Lombardo K.K. Nomura
J.M. Cuta
REFERENCES


### TABLE 1:
TURKEY POINT PWR ORIGEN2 DECAY HEAT PREDICTIONS
AND CALORIMETER MEASUREMENTS

<table>
<thead>
<tr>
<th>Assembly</th>
<th>Burnup (MWd/MTU)</th>
<th>Cooling Time ( t_c ) (d)</th>
<th>Decay Heat (W)</th>
<th>Pred/Meas</th>
</tr>
</thead>
<tbody>
<tr>
<td>D-04</td>
<td>28,430</td>
<td>913</td>
<td>1385</td>
<td>1555</td>
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<td>637</td>
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### TABLE 2:
SAN ONOFRE PWR ORIGEN2 DECAY HEAT PREDICTIONS AND CALORIMETER MEASUREMENTS

<table>
<thead>
<tr>
<th>Assembly</th>
<th>Burnup (MWd/MTU)</th>
<th>Cooling Time $t_c$ (d)</th>
<th>Decay Heat (W)</th>
<th>Pred/Meas</th>
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</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>Meas</td>
<td>Pred</td>
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<tr>
<td>C-01</td>
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<td>3011</td>
<td>359</td>
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<td>C-16</td>
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<td>2012</td>
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<td>Assembly</td>
<td>Burnup  (MWd/MTU)</td>
<td>Cooling Time $t_c$ (d)</td>
<td>Decay Heat (W)</td>
<td>Meas</td>
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<td>----------</td>
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<tr>
<td>C-52</td>
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<td>874</td>
<td>851.4</td>
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**TABLE 4:**
COOPER AND DRESDEN BWR ORIGEN2 DECAY HEAT PREDICTIONS
AND CALORIMETER MEASUREMENTS

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<tr>
<th>Assembly</th>
<th>Burnup (MWd/MTU)</th>
<th>Cooling Time $\tau_c$ (d)</th>
<th>Decay Heat (W)</th>
<th>Meas</th>
<th>Pred</th>
<th>Pred/Meas</th>
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<td>Storage Medium</td>
<td>Contact Conductance (Btu/s-ft-°F)</td>
<td>Temperature Differences (°C)</td>
<td></td>
<td></td>
<td></td>
<td></td>
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<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Rod-to-Rod</td>
<td>Rod-to-Wall</td>
<td>Peak Clad to Boundary</td>
<td>Clad-Canister</td>
<td>Canister-Fuel Tube</td>
<td>Fuel Tube to Boundary</td>
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<tr>
<td>air</td>
<td>no contact</td>
<td>no contact</td>
<td>229 (412°F)</td>
<td>40 (72°F)</td>
<td>59 (107°F)</td>
<td>130 (235°F)</td>
</tr>
<tr>
<td>air</td>
<td>1.0 (10^{-4})</td>
<td>1.0 (10^{-4})</td>
<td>213 (383°F)</td>
<td>24 (44°F)</td>
<td>58 (105°F)</td>
<td>130 (235°F)</td>
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<tr>
<td>helium</td>
<td>no contact</td>
<td>no contact</td>
<td>73 (132°F)</td>
<td>17 (31°F)</td>
<td>14 (26°F)</td>
<td>43 (77°F)</td>
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<tr>
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<td>1.0 (10^{-4})</td>
<td>71 (126°F)</td>
<td>14 (26°F)</td>
<td>14 (26°F)</td>
<td>43 (77°F)</td>
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</table>
### TABLE 6:
EFFECTS OF CONTACT CONDUCTANCE: ECCENTRIC RODS IN ECCENTRIC CANISTER

<table>
<thead>
<tr>
<th>Storage Medium</th>
<th>Contact Conductance (Btu/s·ft·°F)</th>
<th>Temperature Differences (°C)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Rod-to-Rod</td>
<td>Rod-to-Wall</td>
</tr>
<tr>
<td>air</td>
<td>no contact</td>
<td>no contact</td>
</tr>
<tr>
<td>air</td>
<td>1.0 \times 10^{-4}</td>
<td>1.0 \times 10^{-4}</td>
</tr>
<tr>
<td>helium</td>
<td>no contact</td>
<td>no contact</td>
</tr>
<tr>
<td>helium</td>
<td>1.0 \times 10^{-4}</td>
<td>1.0 \times 10^{-4}</td>
</tr>
</tbody>
</table>
CLOSURE LID THERMOCOUPLES (TOP TOTAL)
16 ON LID PLATES / IN EACH THERMOCOUPLE TUBE

EVACUATION AND GAS BAGGUP SYSTEM
CONNECTION PIPE

FUEL ASSEMBLY THERMOCOUPLE TUBES (32)
INSTALLED INTO CONTROL ROOD TUBES

CLOSURE LID THERMOCOUPLES (TOP TOTAL)
6 ON LID PLATE / IN EACH THERMOCOUPLE TUBE

FUEL ASSEMBLY THERMOCOUPLE TUBES (32)
15 ON SUPPORT RING, 30 ON CANISTER BODY
1 ON END CAP

FUEL ASSEMBLY TOP PLATE

CLOSURE LID COPPER GASKET SEAL

Liner

Canister Body

Band Heater (34)

Band Heater Electrical Connections (Wired to Piping)

CLOSURE LID ASSEMBLY

INSULATION SHEATH (12 SECTIONS)

INSULATION BLANKET (LOWER HALF OF LINER ONLY)

TEST ARRANGEMENT IN TEST STAND
WITH SEISMIC RESTRAINT FIXTURE

FIGURE 1:
EMAD Single Unconsolidated PWR Assembly Test Section
(from Unterzuber, 1981)
FIGURE 2:
Cross-Section of EMAD Single Unconsolidated PWR Assembly Test Section
(from Unterzuber, 1981)
FIGURE 3:
Axial View of Liner, Canister, and Fuel Assembly
(T/C designates thermocouple locations)
(from Unterzuber, 1981)
FIGURE 4:
COBRA Temperature Predictions and Experimental Vacuum, Air, and Helium Temperature Data
FIGURE 5:
HYDRA Temperature Predictions and Experimental Vacuum, Air, and Helium Temperature Data
FIGURE 6: Assumed Geometry for Single Assembly Consolidated and Unconsolidated Spent Fuel Storage System
BOUNDARY WALL
121.1°C (250°F)

FUEL TUBE

ROD ARRAY

ACTIVE LENGTH 158"
144"

FIGURE 7:
Axial View of COBRA Model of Fuel Assembly
FIGURE 8: Assumed Cross-Sections for Consolidated Assembly Geometry; (a) Uniformly Axisymmetric, (b) ECCentric Rods in Axisymmetric Canister, (c) ECCentric Rods in ECCentric Canister.
FIGURE 9:
Comparison of Radial Temperature Drop in Air for Axisymmetric Consolidated and Unconsolidated Assemblies (no contact conductance)
FIGURE 10:
Comparison of Radial Temperature Drop in Helium for Axisymmetric Consolidated and Unconsolidated Assemblies (no contact conductance)
FIGURE 11:
Effect of Air and Helium on Radial Temperature Drop in Axisymmetric Consolidated Assembly at 2 kW (no contact conductance)
Comparison of Radial Temperature Drop in Air for Axisymmetric Consolidated and Unconsolidated Assemblies at 1 kW (no contact conductance)
FIGURE 11:
Comparison of Radial Temperature Drop In Helium for Axisymmetric Consolidated and Unconsolidated Assemblies at 1 kW (no contact conductance)
FIGURE 14:
Effect of Rod-to-Rod Contact Conductance on Radial Temperature Drop in Axisymmetric Consolidated Assembly with Air
FIGURE 15:
Comparison of Radial Temperature Profiles in Air for Consolidated Axisymmetric Rods and Eccentric Rods in an Axisymmetric Canister (no contact conductance)
FIGURE 16:
Comparison of Radial Temperature Profiles in Helium for Consolidated Axisymmetric Rods and Eccentric Rods in an Axisymmetric Canister (no contact conductance)
FIGURE 17:
Comparison of Radial Temperature Profiles in Air for Consolidated Axisymmetric Rods and Canister and Fully Eccentric Rods and Canister at 2 kW (no contact conductance)
FIGURE 18:
Comparison of Radial Temperature Profiles in Helium for Consolidated Axisymmetric Rods and Canister and Fully Eccentric Rods and Canister at 2 kW (no contact conductance)
**FUEL SPECS**
- $R_U \approx 27 \text{ GWD/MT}$
- $t_c \approx 1 \text{ yr}$
- $Q_F \approx 1 \text{ kW/ASSEMBLY}$

**FIGURE 20:**
CASTOR-1C CASK CROSS SECTION
FIGURE 21:
Predicted and Measured CASTOR-1C Cask
Axial Temperature Profiles
FIGURE 22:
Predicted and Measured CASTOR-1C Cask
Radial Temperature Profiles
FIGURE 23:
REA (Ridihalgh, Eggers and Associates)
BWR Cask
STAINLESS STEEL

BASKET

ETHYLENE GLYCOL/WATER

LEAD

BWR ASSEMBLIES

0.8 kW CONSOLIDATED AND 0.4 kW UNCONSOLIDATED

FIGURE 24:
REA BWR Cask Cross Section
FIGURE 25:
Predicted REA Cask Axial Temperature Profiles
FIGURE 26:
Predicted REA Cask Radial Temperature Profiles
CANADIAN EXPERIENCE WITH CONCRETE-CANISTER DRY FUEL STORAGE

R.O. Sochaski

Atomic Energy of Canada Limited
Whiteshell Nuclear Research Establishment
Pinawa, Manitoba
ROE 1L0
Canada

ABSTRACT

Concrete canisters have been used at the Whiteshell Nuclear Research Establishment since 1975 October for the storage of used experimental fuel from the WR-1 organic-cooled research reactor. A total of 13 canisters have been built, two of which have been decommissioned, while three are being used for experiments. The remaining canisters store approximately 18 000 kg U in 1320 used fuel bundles. Loading of used fuel continues at a rate of approximately half a canister per year.

After nine years of operating experience, the canisters show no detectable evidence of external or internal deterioration, and routine monitoring of their internal cavities for activity leakage continues to show no positive indication. The results to date confirm the passive features of the concept and the predicted low maintenance and operating costs.
1. INTRODUCTION

The Canadian Concrete Canister Program was initiated in the spring of 1974 to assess concrete canisters as a possible alternative to water pools for the interim storage of used fuel. A detailed concept assessment suggested that concrete canisters would be more economical, would require much less maintenance and surveillance, would retain their structural integrity for a long period of time (>50 years) and could be easily defuelled and decommissioned.

Between 1974 and the first loading of a concrete canister in 1975 October, extensive testing of one cylindrical and one square prismatic concrete canister was undertaken to determine structural integrity, shielding effectiveness, heat-transfer properties, maximum heat loading, longevity, climatic effects, licensability, and to optimize fuel-handling techniques. During the test period, computer codes were developed for determining thermal stress, heat transfer, shielding thicknesses and criticality conditions. It was conclusively shown, by the end of the development program, that this interim used-fuel storage concept was indeed safe, practical and economical. Figure 1 shows the Whiteshell Nuclear Research Establishment (WNRE) site and the location of the concrete-canister demonstration site, near the main laboratory buildings. It also shows the permanent storage site, next to the WNRE Waste Management area, approximately 2 km east of the laboratory buildings.

Shortly after the test program was completed, concrete canisters were licensed and loaded with used CANDU natural uranium, power reactor fuel and used enriched uranium fuel from the WR-1 research reactor, for demonstration purposes. The success of both demonstrations made it easy for WRME to adopt the policy of storing its WR-1 research reactor used fuel in concrete canisters.
Our operating experience to date has been more than gratifying. The passive and rugged qualities of the concept have been confirmed, and so have the economics. A review of our annual operating reports for the nine-year concrete-canister operating period leads us to predict that this interim storage approach will have an excellent future as an economical method for filling future utility storage needs as current water-pool storage capacity is fully utilized.

The following pages will describe, in more detail the background to the AECL concrete-canister program, some design features of the canister, the WNRE approach to fuel handling, our operating experience to date, will provide some cost data, and take a look into the future.

2. BACKGROUND

In spring of 1974 the Whiteshell Nuclear Research Establishment began a development and demonstration program related to the Concrete Canister Dry Fuel Storage Concept (Wasywich et al, 1984). This was undertaken to assess concrete canisters as a possible alternative to water pools for interim storage of used fuel, on the recommendation of the Committee for Assessing Fuel Storage (Morgan, 1977). During 1975 two cylindrical canisters were constructed. One was to be electrically heated and the other was designed for actual fuel storage. Figure 2 shows a typical fuelled cylindrical canister. The following year, two square prismatic canisters were constructed. Again, one was electrically heated and the other was used for fuel storage. The cylindrical canister represented an optimum container design, whereas the square canister was constructed to accommodate the rectangular fuel handling configuration used in CANDU power stations. The electrically heated canisters simulated the expected concrete-canister heat loading, and provided information on structural integrity and heat transfer under various climatic conditions. In 1975 the cylindrical demonstration canister was fuelled with 138 WR-1 used fuel assemblies, which were 500 mm long,
18-element enriched UO$_2$ fuel bundles. In 1976 the program was accelerated, at the request of Ontario Hydro, so they could consider the canister concept for interim fuel storage in 1985. Also, in 1976, the square prismatic demonstration canister was loaded with 360, 500 mm long, 19-element used fuel bundles from the Douglas Point Nuclear Generating Station. In both the above canisters, some fuel bundles, the fuel baskets and the steel canister liner were instrumented with thermocouples to correlate results from the fuelled canisters with those obtained from the electrically heated canisters.

Since the condition of the fuel sheathing was unknown and the concrete-canister concept unproven, the welded carbon-steel baskets containing the fuel bundles were back-filled with an inert gas, to prevent oxidation of the UO$_2$ in case any fuel failures existed, or occurred. The canister lid was seal-welded to the steel canister liner to provide a second containment boundary. Also, the maximum temperature on the hottest bundles in the demonstration canisters was limited to 250$^\circ$C.

Since the construction of the first four electrically heated and fuelled demonstration canisters, nine additional canisters have been constructed at WNRE. Three of these are being used exclusively for experiments, one is being jointly used for an experiment and the storage of WR-1 used fuel, and the remainder are being used for WR-1 used-fuel storage. To date, approximately 1320 fuel bundles (approximately 18 000 kg U) are being stored in these production and demonstration concrete canisters. Figure 3 is a photograph of the concrete canisters at the WNRE Waste Management area, in their loaded configuration.
3. CONCRETE-CANISTER DESIGN

The concrete canister is a hollow, reinforced-concrete cylinder, 2600 mm in diameter and 5200 mm high, standing on a reinforced-concrete cylinder pad (see Figure 2). The cavity is completely encased by a steel pipe and end plates, isolating the interior from the concrete. Thus, the canister is a structurally continuous, reinforced-concrete mass with walls and bottom 890 mm thick. It is fitted with a plug consisting of 25 mm of lead and 762 mm of reinforced concrete.

Each canister is capable of holding six baskets filled with fuel. Several basket designs are available, depending on whether natural or enriched uranium fuel is being stored. Two basket-loading configurations are shown in Figure 4 for enriched uranium used fuel from the WR-1 research reactor. Both baskets are constructed from carbon-steel cylinders 762 mm in diameter, with a welded bottom plate and a central lifting pipe. The top plate is welded on after the fuel has been placed inside. The single-ring basket holds 24 bundles, and the double-ring basket holds 37 bundles. The ring baskets are designed to accept 2.4 wt% U-235 enriched fuel, while ensuring criticality safety. They are capable of withstanding an internal pressure of 100 kPa(g) at 325°C, and provide the first of two containment barriers for the used fuel. The baskets containing the stored fuel can be retrieved at any time.

The canisters at WNRE have been licensed for a maximum internal surface temperature of 150°C, a maximum radial temperature gradient through the concrete wall of 100°C, and a total heat loading of 4.4 kW.
4. USED-FUEL HANDLING

Used fuel discharged from the WR-1 reactor is stored in the reactor water-filled storage bays for a minimum of six months before being considered for transfer to dry storage. When the used fuel is ready for dry storage, it is transferred from the reactor water bay to the WNRE Hot Cells in a flask capable of handling five bundles at a time. Once in the Hot Cells, the fuel bundles are loaded into baskets for safeguards inspection and documentation. The baskets are seal-welded, purged with dry air to remove all moisture, pressure-tested for leaks to 150 kPa(g), then evacuated, purged and back-filled with helium to a pressure of 13.7 kPa(g), and the vent and fill lines crimped and seal-welded. Each basket is then transferred by truck, in the Fuel Basket Transfer Flask (see Figure 5), to the canister site, where the Transfer Flask is placed on top of the concrete canister and the loaded basket lowered into the canister cavity (see Figure 6). Current practice uses a portable A-frame to transfer the Flask from the truck bed to the canister and back again. After six visits, the concrete canister is full and loading of the next canister can be started. Some optimization of used-fuel handling occurred in the late seventies, with the addition of the customized portable A-frame, to ease handling and unloading of the flask at the canister site. Two mobile cranes were used initially to perform this function.

Concrete-canister loading at WNRE proceeds at the rate of approximately one canister every two years.

Although no baskets have been retrieved from the WR-1 production canisters, several baskets have been retrieved from three of our experimental canisters, inspected and reinserted. Since the baskets and retrieval procedures used for the experimental canisters are essentially identical to those used for production units, retrievability is considered to be fully demonstrated.
5. CONCRETE-CANISTER OPERATING EXPERIENCE

WR-1 used fuel, including both UO\textsubscript{2} and UC initially enriched up to 2.4 wt\% U-235, and CANDU natural uranium, UO\textsubscript{2} used fuel from the Douglas Point power reactor, have been stored in concrete canisters at WNRE since 1975. Figure 7 shows the relative positions of the canisters, and their final loading dates. The initial set of four canisters was built near the site parking lot and the WR-1 reactor building, as a public demonstration and for ease of access during the development and testing program, Figure 1). Subsequent canisters have been placed in a compound near our existing site waste management area, approximately 2 km east of the main site buildings.

Each year a report is submitted to our regulatory authority on the operation of the canisters over the previous 12-month period. The report addresses normal and unusual events relating to construction of new canisters and their internals, inspection, fuel handling, movements and loading, site dosimetry, monitoring of liner activity, air sampling, and monitoring of ground and surface water and temperature.

After reviewing nine years of operations reports, the bottom line relating to concrete-canister performance can be summed up by the following quotation taken from several of the reports: "The results of the routine surveillance and monitoring of the WR-1 Spent Fuel Canister Facility confirm another year of satisfactory operation".

The only operational problem with the WNRE canisters during the nine-year period has been ingress of rain water into one of the canisters due to poor sealing of the closure plug. This problem was resolved by covering the tops of all the canisters with roof sealant, and by placing a metal bonnet on each canister, as shown in Figure 3.
Visual inspection of the canister site for the general appearance, integrity of fencing, drainage for run-off water, and general integrity of the concrete canisters is performed on a routine basis. Up to the present, these items have always been found satisfactory.

Gamma radiation field surveys are taken routinely on contact and at 2 m from the canister walls, at 2 m above grade level. Readings have been taken and tabulated over the years and correspond to computer code predictions. In addition, readings to determine the absorbed dose-rate-in-air at the perimeter of the canister site have been taken and tabulated. Once again the measured values correspond to predictions, with nothing unexpected being observed.

Each concrete canister has a drain line and a vent line penetrating the annulus between the sealed baskets and the cavity liner. In addition to draining and venting the annulus, these lines are used for a closed air-circulating system to trap fission products in filters and monitor the space for moisture. The results of the analysis of the air filters over the years have shown no abnormal activity levels. With the exception of the rain-water ingress to one canister annulus early in the program, no further in-leakage has been detected.

Surveillance of ground and surface water for contamination are conducted routinely, and are reported as part of the WNRE environmental monitoring program. No contamination has been detected in either ground or surface water that can be attributed to canister leakage.

Cavity or wall temperatures of canisters loaded with WR-1 used fuel are no longer monitored. The initial four canisters were fully instrumented, and temperature profiles were measured in the cavity and through the canister wall, as a function of cavity power loading. Monitoring of the canisters containing used WR-1 enriched and Douglas Point natural uranium fuel continued for several months after loading, to determine the effects on cavity temperature of seasonal and daily
temperature fluctuations and fuel heat decay. As these tests confirmed the absolute predictability of cavity temperature as a function of fuel heat decay, no provision for temperature monitoring was made for the production canisters.

6. **STORAGE COSTS**

The average cost of a ready-to-load concrete canister, including six baskets, closure plug, all appendages, and construction on a concrete pad at the canister compound is $64 000, in 1984 Canadian dollars. For each canister, the cost of transporting the used fuel from the reactor bay to the Hot Cells, loading the fuel into the baskets, seal-welding, inspecting and transferring the fuel to the canister site for loading is a further $10 000. A fuel transfer flask and a portable gantry cost $26 000 and $85 000, respectively. Site clearing and fencing cost approximately $25 000.

The unit fuel storage cost for the 18 000 kg U of used fuel already stored is given in Table 1 below.
TABLE 1

<table>
<thead>
<tr>
<th>Description</th>
<th>k$/Canister</th>
</tr>
</thead>
<tbody>
<tr>
<td>Canister cost</td>
<td>64</td>
</tr>
<tr>
<td>Transportation and loading costs</td>
<td>10</td>
</tr>
<tr>
<td>Operating and maintenance costs</td>
<td>2</td>
</tr>
<tr>
<td>Fuel basket transfer flask (Annual payments of 3.82 k$, based on PV=26 k$, n=15 years, i=12% and one canister loaded every two years)</td>
<td>8 (rounded)</td>
</tr>
<tr>
<td>Gantry (same financial rules as above)</td>
<td>25 (rounded)</td>
</tr>
<tr>
<td>Site clearing and fencing (for 16 canisters)</td>
<td>2 (rounded)</td>
</tr>
</tbody>
</table>

TOTAL 111

Fuel loading per canister for enriched fuel is 2 330 kg U and fuel loading per canister for natural fuel is 3 780 kg U.

<table>
<thead>
<tr>
<th></th>
<th>Enriched Fuel</th>
<th>Natural Fuel</th>
</tr>
</thead>
<tbody>
<tr>
<td>Unit Fuel Storage Cost</td>
<td>47.60 $/kg U</td>
<td>29.37 $/kg U</td>
</tr>
</tbody>
</table>

7. THE FUTURE

The advantages of the concrete canister storage concept have been apparent to AECL and Canadian nuclear utilities since the early 1970s. The assessment, research, development and demonstration phases during the mid-seventies further reinforced our optimism. The successful storage of WR-1 used fuel during the last nine years has verified that the concept is practical and economical.
From the very beginning, Ontario Hydro has supported AECL in the development of the concrete-canister concept. It has completed several independent concept assessments to determine if it would be economical and practical to meet its own interim fuel storage requirements.

Subsequent to the successful deployment at WNRE and the dissemination of the developmental and operational data to Canadian nuclear utilities and institutions, an effort was made to have foreign nuclear utilities and institutions seriously consider the concept as a practical solution to the imminent shortage of water pool storage capacity for LWR used fuel. Although initially unsuccessful, our involvement in commercializing this concept both within and outside Canada has, within the last year, become marginally successful.

For example, within the past year we have become involved with our sister company, CANDU Operations in Montreal, in assessing the potential of concrete canisters for the storage of G-1 power reactor used fuel. Although it is not yet clear whether the fuel will be stored in concrete canisters or the neighbouring G-2 water pool, the site fuel-handling systems for canning and transporting the fuel will be nearly identical to the existing WNRE concrete-canister used-fuel handling system.

Some conceptual design work was undertaken, under contract from the Commissariat a l’Energie Atomique (CEA), to assess the potential of our concrete-canister concept for the storage of used fuel from the EL-4 power reactor, which will be decommissioned shortly. This work has been completed and a proposal forwarded to CEA for their consideration.

Early this year a license was negotiated between AECL and Nuclear Packaging (NuPac) of Federal Way, Washington, for the use of AECL’s concrete-canister technology to develop a similar concept for the storage of used fuel arising from U.S. light-water reactors. NuPac expects to have the design and approved topical report for their concept
available by the end of calendar year 1985. Marketing is being actively pursued through advertisements and direct contacts, with considerable interest from prospective utilities. Figure 8 shows a concrete cask suitable for 9 PWR or 16 BWR fuel assemblies. Table 2 presents some of the physical parameters for this design, based on normal and heavy concrete construction material.

**TABLE 2**

<table>
<thead>
<tr>
<th></th>
<th>Normal Concrete</th>
<th>Heavy Concrete (Ilmenite)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Dimensions* (mm)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>A</td>
<td>6100</td>
<td>6100</td>
</tr>
<tr>
<td>B</td>
<td>3200</td>
<td>2500</td>
</tr>
<tr>
<td>C</td>
<td>4636</td>
<td>4636</td>
</tr>
<tr>
<td>D</td>
<td>778</td>
<td>778</td>
</tr>
<tr>
<td>Concrete Volume (m³)</td>
<td>46.25</td>
<td>27.14</td>
</tr>
<tr>
<td>Cavity Volume (m³)</td>
<td>2.81</td>
<td>2.81</td>
</tr>
<tr>
<td>Concrete Density (kg/m³)</td>
<td>2413</td>
<td>3500</td>
</tr>
<tr>
<td>Mass Empty (Mg)</td>
<td>112</td>
<td>95</td>
</tr>
<tr>
<td>Mass Loaded (Mg)</td>
<td>118</td>
<td>101</td>
</tr>
<tr>
<td>Number of PWR Fuel Assemblies</td>
<td>9</td>
<td>9</td>
</tr>
<tr>
<td>Number of BWR Fuel Assemblies</td>
<td>16</td>
<td>16</td>
</tr>
<tr>
<td>Fields on Contact (mR/h)</td>
<td>&lt;20</td>
<td>&lt;20</td>
</tr>
<tr>
<td>Poison</td>
<td>Boron</td>
<td>Boron</td>
</tr>
</tbody>
</table>

* Refer to Figure 8 for alphabetical designation of dimensions.

Finally, some development work has been completed on 1/8-scale models of the WNRE canister design to determine if it could be used as transportation container. To date, IAEA drop tests from 9 m, and 1-m puncture tests, have been performed with encouraging results. Although the IAEA standard fire test was not conducted, a numerical simulation of the fire test suggested that there might be problems due to thermal cracking of the concrete. Actual tests would have to be conducted to determine the full extent and severity of thermal cracking. Since drop
testing was conducted on a model of a standard, unmodified concrete canister, it may be that special reinforcing in the impact and thermal-stress areas could improve canister performance under accident conditions, to meet fully the transportation-flask licensing criteria.

8. CONCLUSION

The present WNRE concrete-canister design has turned out to be a practical, passive, safe and economical interim dry fuel storage concept that has fully met all our predictions and expectations. We will add many more canisters to the WNRE site over the years. We expect to see them used for the interim storage of used Canadian power reactor fuel, and we also predict a variation of the concept will be developed in the very near future as a practical and economical solution to the current LWR interim used-fuel storage problem.

REFERENCES

Wasywich, K.M., Chen, J.D., Frost, C.R. and Freire-Canosa, J., 1984 Long-Term CANDU Fuel Integrity in Concrete Canister Storage - Test Results. Atomic Energy of Canada Ltd. and Ontario Hydro, October 18. To be presented at this workshop.

FIGURE 1:
WME Site With Canister Sites Highlighted
FIGURE 2:
Production Concrete Canister
FIGURE 3:

Concrete Canister Farm at the WNRE Waste Management Site
FIGURE 4
BASKET DESIGNS FOR 24 AND 37 BUNDLES
FIGURE 5:
Fuel Basket Transfer Flask
FIGURE 6:

Loading a Basket into a Concrete Canister
CONCRETE-CANISTER SITE LAYOUT

FIGURE 7:
Concrete-Canister Site Layout
FIGURE 8:
Typical for 9 PWR or 16 BWR Elements Concrete Canister
ENGINEERING AND SAFETY FEATURES

OF MODULAR VAULT DRY STORAGE

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GEC Energy Systems Ltd.
Leicester, LE8 3LH, UK

ABSTRACT

This paper discusses the need for interim dry storage and reviews detailed features of the Modular Vault Dry storage concept. The concept meets three basic utility requirements. Firstly, the technology and safety features have been demonstrated on existing plant; secondly, it can be built and licensed in an acceptably short timescale; and thirdly, economic analysis shows that a modular vault dry store is often the cheapest option for interim storage.
1. INTRODUCTION

GEC Energy Systems has engineered a modular vault dry store for irradiated nuclear fuel, which is especially suited to meet the increasing need for interim and longer-term storage. Extensive experience in wet and dry fuel handling and cask and pond engineering (and hence an understanding of the relative merits of the various storage options) has shaped the concept of the store. This paper discusses the engineering and safety features.

The management of the back-end of the fuel cycle is absorbing the attention of the world's nuclear industry, because of the increasing quantities of spent fuel generated by thermal reactors. The escalating cost of reprocessing and the delayed fast reactor programme have resulted in the study of alternative ways of closing the fuel cycle (Reference 1, Deacon, 1982). The basic options following immediate post-irradiation storage are outlined in Figure 1.

Up to the mid 1970s reprocessing was the favoured option, because economic forecasts showed the necessity of establishing an eventual commercial fast reactor programme. The separated plutonium and depleted uranium represented a valuable future energy source, and the value of the plutonium appeared as a credit in the fuel cycle costs. In addition, the highly active waste could easily be conditioned in a form suitable for a final repository.

Despite the availability of cheap oil at that time, countries which either lacked adequate supplies of cheap indigenous fossil fuels, or considered that nuclear energy held the promise of cheap power, adopted a nuclear programme complete with reprocessing, e.g. the UK, France, the USSR, and the USA. Other countries such as West Germany and Japan have made use of the reprocessing facilities in the first four
countries and have also sought to develop their own capability. Canada, with abundant uranium resources, has pursued a policy of long-term storage of its spent fuel.

The subsequent recession in the world's economy and the reduction in demand for power has increased the availability of slightly enriched uranium for thermal reactors at an economic price and has postponed the introduction of fast reactors. As a result, there is no longer a credit for plutonium generated in thermal reactors and recovered in a reprocessing plant; and this imposes the costs of reprocessing on the fuel cycle of the stations in operation. The addition of these reprocessing costs to the fuel costs significantly increases present generating costs with no benefit to the current nuclear power programme. It can be argued that these particular costs should be borne by a future fast reactor programme and not superimposed upon the operational costs of current reactors. It would seem sensible to reserve the reprocessing option by storing the spent fuel compactly and safely, until it can be more clearly seen whether or not the potential fuel value of the plutonium will be realized via the fast reactor option.

The interim storage of spent fuel from thermal reactors not only offers the minimum contribution to the fuel cycle cost, but also preserves the future option of reprocessing. The storage medium can be either liquid (wet storage) or gaseous (dry storage). Wet storage involves an extension of the techniques used in pond and cask designs, where the storage medium is treated water. Different designs for dry storage have used various gases for their storage medium. The advantages offered by dry containment are illustrated by the low cost, demonstrated design and safety features, and demonstrated construction timescales of the Modular Vault Dry Store discussed in this paper.
2. CONCEPT OF THE MODULAR VAULT DRY STORE

The basic concepts of the Modular Vault Dry Store (MVDS) designed by GEC Energy Systems are presented in this Section, and are as discussed previously (Deacon, 1984).

The MVDS design has evolved from the experience gathered in the design, construction and operation of two styles of dry vault store at Wylfa Nuclear Power Station, from the design of the proposed dry store for AGR fuel, and from remote dry fuel-handling experience with a variety of fuel types associated with the UK nuclear programme. Specific MVDS designs have been carried out, for example, for TVA in the USA and TVO in Finland (Deacon, 1982; Maxwell and Deacon, 1982; Davidson and Deacon, 1980).

The design is illustrated in Figures 2 and 3. Figure 2 shows the schematics of the concept and Figure 3 shows a cut-away view of the modular vault dry store. The store is composed of a number of linked identical storage sub-modules, with a reception/dispatch facility located at one end. The other end is designed to allow the facility to be extended with additional modules.

Uncontainerized spent fuel, either in its original or compacted form, is stored within a bank of air-filled, blind-ended tubes. Each tube is plugged and sealed at its upper end and is normally connected into a common manifold system composed of small bore pipework. This system forms a static, sealed, primary containment envelope of high integrity for the stored fuel.

Decay heat from the spent fuel is rejected to the environment by highly reliable passive heat transfer processes. Primary heat is dissipated from the spent fuel to the sealed containment envelope by radiation and convection. Secondary heat rejection from the outside of
the containment envelope to the environment is achieved by self-regulating thermosyphon cooling, using buoyancy-driven ambient air flowing over the outside of the storage tubes. The air is drawn via ducting from the outside of the vault, crosses the tube bank horizontally and exits to the atmosphere via the discharge ducts.

Because the fuel is indirectly cooled, there is no contact between the primary cooling air within the containment envelope and the secondary cooling air discharged to the atmosphere. Inside the storage tubes the air is maintained at a pressure slightly below ambient by an exhaust system, which is connected to the manifold system. This ensures that any possible leakage through the containment envelope is in an inward direction. The flow is nominally zero, and any increase indicates a leak in the containment envelope.

Thus, in addition to the fuel pellet matrix and cladding, there are two engineered barriers against the possible release of radionuclides to the environment, i.e.,

- the containment envelope

and

- the depression of the storage atmosphere.

The diverse nature of these engineered barriers constitutes a high integrity confinement system with a high degree of protection against common-mode failure.

The containment afforded by the storage tubes is surrounded by solid biological shielding that can be designed to any desired degree of radiation attenuation. The structural properties of this arrangement, together with the system of storage tube support, also provides excellent protection against extreme external incidents, such as aircraft crash or seismic disturbance.
Spent fuel is remotely handled into the storage tubes by a fully shielded charge machine.

A typical module can store about 700t of LWR fuel in its original form, but the packing density can be increased if the fuel assemblies are compacted.

The store is composed of a group of storage vault modules and the ease with which further storage modules can be added without disturbance to those existing is a distinct advantage (Figure 3). This allows the system to be closely tailored to individual site storage requirements.

3. **IMPORTANT FEATURES OF THE MODULAR VAULT DRY STORE**

Many aspects of dry interim storage for spent fuel have been discussed in past workshops on this subject. The important features of the Modular Vault Dry Store are as follows:

- Selection of the Storage Environment
- Containerization/Containment Aspects
- Cooling Regimes
- Spent Fuel Handling
- Applicability to Other Fuel Types
- Length of Storage
- Licensability
- Cost

These features are examined in greater detail in the following sections.
3.1 SELECTION OF THE STORAGE ENVIRONMENT

The question of which cover-gas, and the appropriate initial temperature level for the spent fuel has been discussed previously (Wheeler, 1984), and it is worth re-stating the argument.

The selection of the storage medium for the spent fuel has a marked effect on clad and pellet reactions, and it therefore seems natural to select an inert, or relatively inert, cover gas such as helium, argon or carbon dioxide. Although these cover gases reduce the reactions and allow higher storage temperatures for the spent fuel, they can become diluted with air in some fault situations. Therefore, this choice relies upon a high quality containment for the spent fuel that must be assured to the required standards of integrity for a storage period of many decades. Additional monitoring equipment will, as a result, need to be installed to check the pressure and contents of the containment. (In fact, some of the necessary sensors are not yet available but are under development.) Despite these engineering provisions the resulting safety arguments require a probabilistic risk analysis to establish the necessary confidence in the remoteness of any undesirable long-term consequence associated with coincidence of containment penetration and perforated clad.

These considerations do not preclude the use of an inert gas, especially if highly rated fuel must be stored early in its decay history, but it is better to select an initial temperature level that will prevent excessive clad or pellet degradation, if air should gain access to the fuel at any time. All credible fault scenarios for the safety case converge in the ultimate on air as the storage medium, and it is therefore worth considering its use from the outset. Even if this is not always possible then at least a change in cover gas from inert to air at the earliest opportunity should be considered.

If air is used as the storage medium, its oxidizing effect (particularly on the pellets via any clad perforations) will need to be considered. Firstly, the level of production of UO$_2$ aerosols,
the release of fission products from the oxidizing pellet, and perhaps the spalling of pellet fragments must be minimized. Secondly, the products and consequences must be localized and design measures taken to prevent the radioactive material from migrating within the containment and especially from escaping through any leakage pathways in the primary containment.

The first aspect can be covered by selecting an appropriate initial storage temperature for the fuel. There is no point in imposing unnecessarily high initial temperatures (and hence decay heats), on the fuel for any interim store if they can be avoided. (See also Section 3.5.)

Data presented at this workshop, together with reports of recent work (Reference 6) show that, provided the initial temperature is sufficiently low, it should be possible to constrain any pellet oxidation to the $\text{U}_9\text{O}_7$ phase throughout the interim storage period. This in itself should prevent pellet swelling and further defect exacerbation. Application of oxidation rates derived from extrapolating existing data to the much lower temperatures expected for stored fuel would only lead at most to the oxidation of two or three pellets adjacent to the original defect. Even this assumes that the original defect occurs at the point of maximum temperature.

Experience to date indicates that the positions of any initial clad perforations are likely to be random and thus occur at spots where the initial temperatures are lower than the maximum temperature specified for a design. Therefore, oxidation will be small. Because the temperatures are low, the actual release fractions of the fission products within any oxidizing pellet will also be much less than the amounts indicated by high temperature reactor fault situations. The only exception would perhaps be the release of the noble gases.

The localization of consequences and products is also partly covered by the above points, but if air is chosen as the cover gas (as used in the USA Stores - Cells 4 and 5) it is also possible to employ
a secondary diverse containment system. This takes the form of a system to maintain the pressure of the cover gas within the primary containment slightly below ambient. This provides a second and diverse means of retaining any low quantities of radioactive arisings. This system is also capable of controlling and conditioning the quality of the relatively small volume of air within the primary containment providing the cover gas. For example, its moisture content can be monitored and maintained at a suitable dryness level to control any possible moisture-enhanced reactions. This could be desirable if perforated fuel has previously become water-logged in a pond. Because only long-term inhibiting conditions may be required, the drying plant would not need to be designed for continuous, high availability usage. (See also Sections 3.2.1 and 3.2.3.)

Apart from the localized consequences of pellet oxidation mentioned above, storage in air at the temperature envisaged should not lead to structural degradation of the complete fuel assembly. As a result, the complete assembly should be capable of being handled as an integral unit at the end of the interim storage period.

In conclusion, it can be seen that pellet oxidation is the controlling phenomenon from the safety aspect, even if mechanisms are subsequently found that can induce cladding penetrations at temperatures less than 170°C. At this and lower temperature levels, the current oxidation work indicates that the release from any oxidizing pellet to the immediate storage medium will be very low, since only the $\text{U}_3\text{O}_7$ phase is likely to be present for the periods contemplated for interim storage. Because the rates of oxidation are small, release levels and rates will be low and will consist mostly of noble gases. Any particulate and volatile components from the pellet matrix will tend to remain in-situ at these low temperature levels. Therefore, while an inert cover gas solution may seem attractive, air is considered to be the preferred environment because its adoption not only leads to an acceptable safety case, but also bestows many economic and operational advantages.
3.2 CONTAINERIZATION/CONTAINMENT ASPECTS

Many studies for the dry storage of spent fuel have selected an inert gas for the storage medium. To retain this gas some form of sealable bottle or canister has been chosen to encapsulate the fuel. As a result the concept of a canister or can has been taken to be an engineering requirement, and examples can be found in conceptual designs for casks, silos and vaults. In practice the only requirement that is likely to be specified is at least one high integrity containment other than the fuel cladding. This may be provided by a container, a canister, a cask, or the sealable tube used in the MVDS. The containment must be capable of retaining any credible potential release of activity from the fuel to within the national/international radiological release level, and preferably to ALARA. In this respect the containment is no different from the philosophy applied to reactors, where perforated fuel cladding can exist.

The can or canister concept has been a favourite choice in previous studies, but the characteristics of the sealable tube should have received more consideration. To remedy this Table 1 presents a detailed comparison of the tube and canister. The table clearly shows the engineering and safety advantages of the tube concept. The main advantages are summarized in Table 2.

The following sections discuss the most important features of the tube system outlined in Table 1.

3.2.1 The Depression System

Despite the precautions taken to preserve and maintain the integrity of the fuel, a release of radioactivity can be anticipated to the small internal volume of the storage tubes that form the primary containment. This containment boundary is designed to very high standards and can be expected to have the degree of integrity normally associated with quality containment designs in the nuclear industry.
## TABLE 1
### COMPARISON OF THE TUBE AND CANISTER CONCEPTS FOR VAULT STORAGE

<table>
<thead>
<tr>
<th>FUNCTION</th>
<th>TUBE</th>
<th>CANISTER</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Storage Environment</td>
<td>Any gas may be selected as a filling.</td>
<td>Any gas may be selected as a filling.</td>
</tr>
<tr>
<td></td>
<td>Subsequent control is possible over its type, quality, and the injection of inhibitors.</td>
<td>Once selected there is no easy subsequent control over its type, quality, or potential use of inhibitors.</td>
</tr>
<tr>
<td>2. General Containment Characteristics</td>
<td>Provides flexibility in the achievement of customers' requirements for containment.</td>
<td>All radioactive species are contained within the sealed canister (see also 4,6,7 for faulty canisters).</td>
</tr>
<tr>
<td></td>
<td>a) The tubes to be sealed permanently or temporarily with the closure effecting the seal being easily inspectable at a later date. (See also 'c' below.)</td>
<td>In order to encapsulate the fuel in the first place and to inject the chosen storage medium a preparation facility in the form of a large hot-cell will be required. This facility is relatively complicated, and will house remote closure, fuel and canister feed and translation mechanisms, and perhaps a drying bay. The facility will also be used for inspection of spent fuel because canisters will need to be cut open prior to inspection. This air-filled facility contains any potential release by means of a depression and filter system.</td>
</tr>
<tr>
<td></td>
<td>b) The tubes to be fitted with a manifold pipework system, itself connected to a high efficiency filter/exhauster system. This arrangement confers the ability to</td>
<td></td>
</tr>
<tr>
<td></td>
<td>i) have a reduced pressure inside the tubes giving a diverse secondary containment system</td>
<td></td>
</tr>
<tr>
<td></td>
<td>ii) have a means of continually or intermittently checking the containment integrity</td>
<td></td>
</tr>
<tr>
<td></td>
<td>iii) monitor the gaseous storage environment</td>
<td></td>
</tr>
<tr>
<td></td>
<td>iv) purge or inject fresh environments</td>
<td></td>
</tr>
</tbody>
</table>
Noble gases can be trapped and stored should this be deemed necessary (NB, if no measures are taken to restrict noble gas release then the amounts are comparable to, or less than, a reactor and orders of magnitude less than for a reprocessing plant).

c) The system to be sealed with the exhaust system isolated. In this mode the performance is the same as a canister with the added advantage that control of the environment type, quality, etc., is easily achievable.

A small preparation facility contained by a depression/filter system would only be necessary if the fuel ever required to be dried.

The charge machine will need to be considered a containment – a requirement less onerous than for reactor refuelling machinery.

<table>
<thead>
<tr>
<th>FUNCTION</th>
<th>TUBE</th>
<th>CANISTER</th>
</tr>
</thead>
<tbody>
<tr>
<td>3. Containment Complexity</td>
<td>Simple carbon-steel tubes with simple inspectable elastomer closure seals. More complicated pipework and depression/filter system (if fitted). The head-end preparation cell (if required) facility is simpler and of smaller volume.</td>
<td>Simple carbon-steel cans but with very high integrity closure welds. All closure functions need to be carried out remotely. The preparation facility is a large volume, multi-penetration hot-cell.</td>
</tr>
<tr>
<td>FUNCTION</td>
<td>TUBE</td>
<td>CANISTER</td>
</tr>
<tr>
<td>-------------------------------</td>
<td>----------------------------------------------------------------------</td>
<td>--------------------------------------------------------------------------</td>
</tr>
<tr>
<td>4. Containment Integrity</td>
<td>A second diverse containment barrier can be provided by the use of the depression/filter system.</td>
<td>Canisters rely upon the quality of the final closure and also to a lesser extent on the quality of the storage environment.</td>
</tr>
<tr>
<td></td>
<td>Integrity of the primary containments can be verified by means of the depression system. Location and magnitude of any fault can be obtained easily. No leakage to environment occurs during identification/rectification process.</td>
<td>Undisclosed faults in the canisters cannot be discounted. These will be difficult to find and rectify. Therefore there is no direct control of potential release to the environment in these circumstances. Release to atmosphere will continue until the canister is identified and removed.</td>
</tr>
<tr>
<td></td>
<td>The preparation cell (if required) relies on the integrity of the depression filter system.</td>
<td>The preparation facility relies on the integrity of the depression filter system.</td>
</tr>
<tr>
<td>5. Circuit Contamination of Associated Systems</td>
<td>Head-end facility (if required) of much smaller volume will become contaminated. The charge machine and charge face could become contaminated but much less-so than with equivalent style GCR refuelling machines.</td>
<td>The large volume preparation facility will become contaminated. The charge machine will remain clean since it handles containerized fuel, except for faulty containers.</td>
</tr>
<tr>
<td>6. Monitoring</td>
<td>It is possible to continuously monitor the internal contents of individual, groups, or the bulk of the tubes on a continuous or intermittent basis, giving early warning of any degrading fuel. Also, bulk sampling of the exit thermosyphon cooling is carried out.</td>
<td>Generally, only bulk sampling of the exit air and the air external to the canister. Internal monitoring of the canisters is only possible with great difficulty and will require the development of special sensors etc.</td>
</tr>
<tr>
<td>FUNCTION</td>
<td>TUBE</td>
<td>CANISTER</td>
</tr>
<tr>
<td>---------------------------</td>
<td>-----------------------------------------------</td>
<td>--------------------------------------------------------------------------</td>
</tr>
<tr>
<td>7. Inspection</td>
<td>The fuel may be easily inspection at any time using the charge machine. This is an important advantage for</td>
<td>This requires retrieval of the canisters followed by a cutting procedure in the prep-cell, followed by rewelding. This is a clumsy, inflexible and costly procedure.</td>
</tr>
<tr>
<td></td>
<td>i) verification purposes</td>
<td></td>
</tr>
<tr>
<td></td>
<td>ii) long-term examination of fuel for degradation</td>
<td></td>
</tr>
<tr>
<td></td>
<td>The tube can be easily withdrawn for inspection.</td>
<td></td>
</tr>
<tr>
<td>8. Functional Requirements</td>
<td>Only nominal anti-corrosion thickness required. Permanently installed (although removable if desired) to form a structure and guidance for the cooling airflow.</td>
<td>Canisters must be thick to withstand long-term internal and external corrosion. Also, they must be capable of being dropped and not rupturing. Internal structures are required to support the stack of cans.</td>
</tr>
<tr>
<td>i) Corrosion</td>
<td></td>
<td></td>
</tr>
<tr>
<td>ii) Dropping</td>
<td></td>
<td></td>
</tr>
<tr>
<td>iii) Structural function</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>If the fuel is dropped, it is safely contained within the tube without imposing axial loads on the tubes.</td>
<td></td>
</tr>
<tr>
<td>9. Retrieval of Damaged Fuel/Container</td>
<td>Fuel transferred within the charge machine to a new location and the offending tube replaced with a new tube using the charge machine.</td>
<td>If a canister or fuel is known to be faulty, it needs to be transferred to the head-end preparation cell for rectification i.e. transfer to a new container.</td>
</tr>
<tr>
<td>10. Ancillary Requirements</td>
<td>May only require a simple preparation cell.</td>
<td>Requires a prep-cell to close and test the canisters and also to rectify faulty canisters.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>This facility will be required throughout the store lifetime.</td>
</tr>
<tr>
<td>11. Replaceability</td>
<td>Tubes are replaceable with relative ease.</td>
<td>Canisters are replaceable with ease.</td>
</tr>
<tr>
<td>FUNCTION</td>
<td>TUBE</td>
<td>CANISTER</td>
</tr>
<tr>
<td>-----------------------</td>
<td>----------------------------------------------------------------------</td>
<td>--------------------------------------------------------------------------</td>
</tr>
<tr>
<td>12. Flexibility</td>
<td>Flexible, can easily add further capacity as required in a modular system.</td>
<td>Ditto as for tube.</td>
</tr>
<tr>
<td>13. Operability</td>
<td>Simple and easy to maintain working parts. All mechanisms located outside the containment boundary. Minimum of staff required.</td>
<td>The preparation cell contains mechanisms inside the containment boundary. This relatively complicated facility which is a necessary requirement for a canister concept will require more staff to operate and maintain.</td>
</tr>
<tr>
<td>14. Waste Generation</td>
<td>The tubes are re-usable.</td>
<td>Canisters are not re-usable.</td>
</tr>
<tr>
<td></td>
<td>The depression system filters will not normally be subjected to flow and a long life for the filters can be expected.</td>
<td>Rejected, and subsequent faulty canisters that do not meet the containment acceptance criteria will also form a waste stream.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Canisters housing fuel removed for periodic inspection will also add to the waste.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>The preparation cell H&amp;V filters will represent, or give rise to, waste whether replaced or regenerated.</td>
</tr>
<tr>
<td>15. Subsequent Fuel Removal</td>
<td>Fuel is easily removable and can be transported dry and without an overpack. This is the most acceptable form to a reprocessor (if this is required).</td>
<td>Unless the fuel is removed from its canister, the canisters will represent an undesirable waste stream to a reprocessor.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>NB 1) Containerized fuel might make final lifting easier, but since gross degradation of fuel assemblies is not expected, this is a doubtful advantage.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2) It is questionable whether original vault storage canister could be used directly as an overpack for a &quot;heritage&quot; store or final repository.</td>
</tr>
<tr>
<td>FUNCTION</td>
<td>TUBE</td>
<td>CANISTER</td>
</tr>
<tr>
<td>--------------------------------</td>
<td>----------------------------------------------------------------------</td>
<td>--------------------------------------------------------------------------</td>
</tr>
<tr>
<td>16. Decontamination and</td>
<td>Less material since the tubes are thin.</td>
<td>More material that requires eventual disposal since large volume of canister material.</td>
</tr>
<tr>
<td>Decommission</td>
<td>Complete removal of in-vault structures when the tubes are withdrawn.</td>
<td>In-vault structures will require disposal and could possibly be contaminated.</td>
</tr>
<tr>
<td></td>
<td>The charge machine will require decontamination.</td>
<td>Preparation cell will require decontamination and disposal of the in-cell equipment.</td>
</tr>
<tr>
<td></td>
<td>The pipework system (if installed) will require final decontamination prior to decommissioning.</td>
<td>The charge machine will not require decontamination.</td>
</tr>
</tbody>
</table>
TABLE 2

MAIN ADVANTAGES OF SEALABLE STORAGE TUBES

<table>
<thead>
<tr>
<th></th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Any gas may be selected as the storage medium.</td>
</tr>
<tr>
<td>2</td>
<td>Tubes sealed permanently with a replaceable/inspectable plug/seal.</td>
</tr>
<tr>
<td>3</td>
<td>Simple carbon-steel tubes, with protective coating on the outside.</td>
</tr>
<tr>
<td>4</td>
<td>Can be fitted with a diverse secondary containment system, i.e. a system to produce a pressure inside the tubes slightly less than ambient.</td>
</tr>
<tr>
<td>5</td>
<td>Can continuously monitor the internal contents of individual and groups of tubes.</td>
</tr>
<tr>
<td>6</td>
<td>Can determine the integrity of individual tubes and the bulk containment.</td>
</tr>
<tr>
<td>7</td>
<td>Can alter or condition the storage medium.</td>
</tr>
<tr>
<td>8</td>
<td>The fuel can be easily inspected for verification or other reason.</td>
</tr>
<tr>
<td>9</td>
<td>The tubes can also be withdrawn for inspection or refurbishment/replacement.</td>
</tr>
<tr>
<td>10</td>
<td>Less material for eventual decontamination and decommissioning.</td>
</tr>
</tbody>
</table>
To further enhance containment integrity, all the storage tubes can be maintained at a pressure slightly less than atmospheric. This engineered safety has been successfully employed and licensed on the existing dry stores (Cells 4 and 5) at Wylfa power station (Maxwell and Deacon, 1982; Ealing, 1983). The depression is achieved by means of a small-bore high-integrity pipework system connected both to the individual storage tubes and to redundant small capacity fans, which exhaust to atmosphere via conventional particulate filters. This diverse protective system prevents any escape of radioactivity, since any potential leakage will be inward. The system also provides a continuous monitor on the integrity of the complete primary containment boundary, because there will be no airflow from the storage tubes unless a leak should happen to develop.

The depression system is not required to function on essential supplies, because coincidence of depression system failure and perforation of the primary containment will be very unlikely, due to the engineered integrity of the primary containment.

3.2.2 The Individual Storage Tubes

Because the storage tubes can be isolated individually, the products of any fuel degradation can be localized.

The storage tubes are designed for long life and protected on their outer surface against corrosion from the ambient air of the thermosyphon cooling system. It is, however, possible for them to be removed via the charge machine for inspection or replacement, if this ever becomes necessary.

The use of individual storage tubes allows a rapid installation programme during initial construction. Also, the installation of the tubes within a sub-module can be delayed to minimize the capital investment, if the potential storage capacity of a vault module is not initially fully utilized.
3.2.3 Control of the Storage Environment

Each storage tube is capable of being isolated, and it is possible to monitor the storage medium within for its quality and also for any radioactive release. This may be done on a continuous or intermittent basis.

3.2.4 Fuel Inspection

The charge machine is capable of incorporating the necessary inspection and monitoring equipment so that the fuel assemblies in the tubes can be inspected when desired.

Remote viewing inspection equipment on the machine can visually examine any chosen assembly at any time, simply by visiting the chosen storage tube and hoisting the assembly into the machine. Gamma-ray spectroscopy can also be installed to gain some idea of radioactive inventory and hence irradiation history, burn-up etc. This facility can also provide individual characterization for each fuel assembly.

3.3 COOLING

Most concepts for vault storage have opted for a cooling system that uses air as the cooling medium to remove the bulk decay heat from the store. A thermosyphon system driven by natural buoyancy forces is to be preferred, because it is entirely passive with intrinsically reliable cooling mechanisms, and the cooling medium can never be lost. The draught of cooling air can be either vertical, or cross-flow.

The vertical type is often selected, partly because of the tendency for heated coolants to rise, and partly because of past reactor experience with vertical cooling.
Cross-flow thermosyphon air-cooling (see Figure 2) has been adopted by GEC Energy Systems in preference to vertical cooling, because of the benefits it bestows, the chief being:

- This mode of air-cooling has already been shown to be effective in the original vault stores that were built into Wylfa power station (see Figure 4).

In addition, a further understanding of the cooling processes at Wylfa has been obtained from analysis and measurements.

- The flow through the vault is more uniform, with predictable heat transfer regimes and correlations.

- There are no internal structures to delineate the airflow, thus enabling a higher storage density and cost reductions to be achieved.

- The volume of the storage vault is used more efficiently, because a tu plenum for the exit cooling air is not required.

- The warm exit air does not flow across the underside of the charge face and thermal and structural effects in the charge-face structure are therefore reduced.

- The air resistance of the tube array is smaller for cross-flow, enabling a greater flow and hence lower temperatures for a given heat load in the store.

A 1/4 scale model of one storage module quadrant has been constructed (see Reference 6; Ealing, 1983) to extend the heat transfer and fluid flow correlations, and to demonstrate the absence of re-circulation at full and partial heat loadings on the quadrant. Test work is now complete, and the results provide full confidence for the viability of this simple cooling mode.
The heat transfer research programme has included full scale testing of a PWR fuel assembly housed within a representative tube. This programme (outlined in Reference 6; Ealing, 1983) has enabled the necessary radial and axial, pin and air temperature profiles to be explored over a representative range of decay heats (electrically simulated) for a set of imposed tube temperatures. The results show close correlation with computer predictions and enable a detailed understanding of the internal flows and heat transfer mechanisms within a tube.

3.4 FUEL HANDLING

Most of the world's experience with spent fuel handling has been undertaken with the operator having direct visual control of the activity. Vault storage requires blind fuel handling and, as a result, a degree of apprehension has been expressed in some quarters. This apprehension is completely misplaced, because a wealth of design, manufacturing and operating experience has been gained in the UK, where fuel has been handled blind and dry almost without exception since the advent of nuclear power. The remote latching and transfer (some of it at high speed) of new and spent fuel, under automatic sequence control, has been accomplished with an exemplary record for safety and availability (Maxwell and Deacon, 1982). Figures 5, 6 and 7 illustrate the style of machinery involved.

This experience has enabled the confident production of generic designs of dry and remote fuel-handling machines designed to transfer spent fuel between reactor ponds, casks and all types of dry storage facilities.
3.5 **APPLICABILITY TO VARIOUS SPENT FUEL TYPES**

The design is not restricted to any one type of fuel. The design and development has been carried out for a range of various types of fuel and vitrified waste, and satisfactory operating temperatures for all types of fuel has been demonstrated. The varying temperature requirements for various spent fuel types can be accommodated by two methods.

Firstly, the temperature of stored fuel can be controlled by selecting fuel for interim storage with the appropriate level of decay heat. There is no point in imposing unnecessarily high initial temperatures (and decay heats), on the fuel for any interim store if they can be avoided.

This condition can be achieved by retaining the spent fuel in the immediate post-reactor stores as long as possible, to allow the heat and radiation levels to reduce (within the constraints imposed by the ultimate capacity of the stores). This approach is not only beneficial for interim storage itself, but also aids the design of casks to transfer fuel from the reactor store.

As an example, the initial temperature limit for PWR fuel can be achieved with fuel assembly decay heats of about 1 kW; this level being reached after about 4-1/2 years in the reactor ponds. Fuel with a higher decay heat rating can be prohibited from insertion into the store by a detector system (embodying generous redundancy and mounted on the charge machine) that monitors the level of decay heat. (Such a system is currently in use at Wylfa.)

Secondly, the temperature of the spent fuel contained within the storage tubes (in its original or compacted form) is determined not only by the decay heat rating and geometry of the fuel, but also by the temperature of the storage tubes.
This latter temperature is governed by the total quadrant heat load and by the cross-flow cooling induced by the thermosyphon system, which is determined by the geometry of the vault. The chief parameters of interest are:

- The height of the exit ducts.
- The number of tubes and hence heat generation.
- The arrangement of the tubes, i.e. the number of tube banks and the height and diameter of the tubes.
- The tube pitch/diameter ratio and whether the tubes are on a square or triangular pitch.
- The resistance to the cooling airflow. This is largely a function of the tortuous path required for shielding purposes, plus the resistance of the tube bank.
- The ambient air conditions. These are imposed by local meteorology. High summer values are used to assess the store and fuel performance.

The flexibility of the design is such that variations and combinations of the above parameters can be accommodated to vary tube and hence fuel temperatures within a few tens of degrees centigrade. For example, with a given vault geometry, an increase in duct height from 15 m to 30 m will give a decrease of about 8°C in fuel temperature.

3.6 LICENSEABILITY

The basic safety features of the MVDS design (containment, cooling and shielding) are the same as used on the existing UK dry storage installations, which have been licensed by HM Nuclear...
Installations Inspectorate. Safeguards provisions can easily be met and re-verification of the content of the store can be achieved directly using the charge machine.

3.7 ECONOMIC CONSIDERATIONS

The MVDS design is available in several configurations for various applications, either at the reactor or away from the reactor. A typical storage module of the arrangement illustrated in Figure 3 will store 400tU unconsolidated PWR fuel or 660tU BWR fuel with a land area usage of 0.6tU/m². This efficient utilization of space, and the modular arrangement of the design, allows the design to be matched to small local interim stores, as well as very large central stores. The charge machine, storage tubes and charge face structure (vault roof) can be factory manufactured and installed after completion of the civil works, allowing rapid construction timescales. A significant factor in reducing the operating and capital costs of the MVDS for fuel storage is the avoidance of hot-cell facilities to remotely carry out fuel containerization and routine fuel inspection during the storage period.

Capital and operating cost estimates are given in Tables 3 and 4 for the MVDS. These are based on reactor site storage for a twin-unit European BWR station and the estimates include complete design, manufacture, construction, installation and commissioning of MVDS facilities, plus reception and handling facilities in the store for a transfer cask system. The estimates also include owner's interest during construction. The fuel assemblies are assumed to be stored intact, although storage of consolidated fuel is feasible after a suitable cooling period, allowing a greater storage capacity.

These estimates (see also Figure 8) are based on an exchange rate of $1.5 to the pound sterling for comparison with other data and they are considered to be relatively insensitive to the location (country/site).
Modularization is an important aspect of the economics of the MVDS. Modules as small as 200tU can be constructed, although in the case described above 700tU modules (including a margin for irradiated core components) gave the best economic and technical solution. Figure 8 shows estimated capital costs of incrementally built MVDS storage modules for BWR spent fuel.

A programme of five years is foreseen from order to commissioning, including two years before start of works on site. This programme is supported by GEC Energy Systems' experience of building two 350tU additional vault stores at Wylfa in less than three years each, from date of order.

**TABLE 3**

ESTIMATED COMMERCIAL CAPITAL COSTS OF MVDS FOR 1300tU BWR FUEL IN $THOUSAND AT 1984 PRICES

<table>
<thead>
<tr>
<th>Description</th>
<th>Cost</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design, project QA software</td>
<td>3,400</td>
</tr>
<tr>
<td>Capital (civil, mechanical, electrical, etc.)</td>
<td>18,400</td>
</tr>
<tr>
<td>Construction and Installation</td>
<td>3,900</td>
</tr>
<tr>
<td>Owners' interest during construction</td>
<td>5,400</td>
</tr>
<tr>
<td><strong>Total capital cost</strong></td>
<td>31,100</td>
</tr>
<tr>
<td><strong>Unit capital cost, $/kgU</strong></td>
<td>24</td>
</tr>
</tbody>
</table>

**TABLE 4**

ANNUAL OPERATING COST OF MVDS FOR 1300tU FUEL IN $1984

<table>
<thead>
<tr>
<th>Description</th>
<th>Cost</th>
</tr>
</thead>
<tbody>
<tr>
<td>Personnel</td>
<td>128,000</td>
</tr>
<tr>
<td>Power</td>
<td>47,000</td>
</tr>
<tr>
<td>Maintenance</td>
<td>150,000</td>
</tr>
<tr>
<td>Contingency</td>
<td>30,000</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td>355,000</td>
</tr>
</tbody>
</table>
An independent survey of dry store costs has recently been published in Reference 8, and the summary figure (Figure 9 in this paper) shows the clear advantage enjoyed by the vault concept (Deacon, 1984).

3.8 SIZE OF INSTALLATION

A factor that is not always appreciated is the small size of a vault dry store compared to the size of the station. The compact nature of the installation is vividly shown in Figure 10, which illustrates a MVDS for the 30 year output from a 2 x 1000 MWe LWR station.

4. CONCLUSIONS

This paper has described the important operating principles and safety features of a modular vault dry store with tube containment. The system is safe and incorporates demonstrated technology. Moreover, it meets the three basic requirements of potential owners.

First The technology and safety features have been demonstrated and licensed on similar existing plant.

Second It can be built and licensed on an acceptably short-time scale.

This has been demonstrated by the performance actually achieved on Wylfa Cells 4 and 5 (Figure 11).

Third The economics of the concept are such that the vault represents one of the best options for interim storage.
ACKNOWLEDGEMENTS

The authors wish to acknowledge the work and advice given by many colleagues and to thank the Directors of GEC Energy Systems for permission to publish this paper.
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Storage,  
FIGURE 1  Basic options for the back-end of the fuel cycle.
FIGURE 2  Schematic of Modular Vault Dry Storage system, illustrating spent fuel containment systems and thermosyphon cooling flow.
1. Spent Fuel or Vitrified Waste Receipt/Dispatch Building
2. Charge Hall
3. Charge Machine and Gantry
4. Storage Sub-Module fully equipped with Storage Tubes and loaded with Spent Fuel or Vitrified Waste
5. Storage Sub-Module prior to installation of Storage Tubes
6. Storage Sub-Module fully equipped with Storage Tubes and partially loaded with Spent Fuel or Vitrified Waste
7. Air filled Storage Tubes Primary Containment for Spent Fuel or Vitrified Waste
8. Thermosyphon Cooling Air Inlet Duct
9. Cooling Air Exit Duct
10. Offices
11. Transport Cask and Transporter

FIGURE 3 Cutaway view of Modular Vault Dry Store.
FIGURE 4  Wylfa dry storage system: Cells 1, 2 and 3.
FIGURE 5  Charge face refuelling machine at Wylfa.
FIGURE 6 General view of transfer machine and trolley at Wylfa.
FIGURE 7 Hartlepool/Heysham fuelling machine.
FIGURE 8 Estimated capital costs of modular vault dry storage for BWR spent fuel, built incrementally in 650tU modules.
FIGURE 9    Cost ranges of wet and dry storage options for spent fuel.
FIGURE 10 Relative size of a Modular Vault Dry Store at a 2000MW LWR power station.
FIGURE 11 Time schedule for planning and construction of Wylfa Cells 4 and 5.
PROGRESS ON DRY STORAGE OF IRRADIATED FUEL IN THE FRG

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ABSTRACT

AFR dry storage in transport/storage casks is a licensed technology in the FRG. Construction of the first 1,500 MTU facility is completed and commissioned for operational use in 1984. A second AFR cask storage facility of the same capacity is under construction since July 1984.

Full scale dry storage cask demonstrations with irradiated PWR and BWR fuel and hot cell tests have been conducted since 1979 and support the technical bases for dry storage of spent fuel. Based on the operational experience with cask storage, dry storage can be considered a proven technology.
1. **INTRODUCTION**

The back-end fuel cycle strategy in the FRG is based on interim storage and subsequent reprocessing of spent fuel. The waste shall be disposed of in a geologic repository. Parallel, alternative back-end fuel cycle techniques such as the direct disposal of the spent fuel without reprocessing are being investigated and might at a later stage be used for the final disposal of spent fuel, which is not suitable for reprocessing.

Based on the estimated accumulation of spent fuel by the end of this century the German back-end fuel cycle requirements will be met in two ways, i.e. reprocessing in a German reprocessing plant with an annual capacity of 500 MTU, for which the licensing procedure is under way since 1982 and the first construction license is expected for 1985, and reprocessing at the French and British facilities of COGEMA and BNFL. Therefore, temporary storage capacity in nuclear power plant pools and additional AFR storage capacity plays an important role in spent fuel management in the FRG.

This paper describes the actual status of licensing and operating of AFR dry storage in transport/storage casks and presents results of licensed technical scale demonstrations.

2. **STATUS OF AFR DRY STORAGE IN CASKS**

Dry interim storage in transport/storage casks is a licensed technology in the FRG. Several types of CASTOR series casks for both storage and transportation have already been licenced. Additional licenses for casks including e.g. the TN 1300 cask are expected in the near future.
The actual status of licensing and construction of AFR interim storage plants at different sites in the FRG can be described as follows:

- Construction of the first 1,500 MTU AFR-facility at Gorleben was started in February 1982. Construction of the facility including a storage building for low-level waste is completed, the operating license has been issued. Shipment of the first casks is expected in the near future.

- License application for a second 1,500 MTU facility was filed in October 1979 for the Ahaus site. The construction license was issued in October 1983. Site construction work has started in July 1984, the facility is scheduled for completion in the last quarter of 1986.

- In combination with the German reprocessing plant at Wackersdorf or Dragahm licenses where applied in 1982 and 1983 for a 1,500 MTU and 500 MTU facility, respectively. A final decision and construction license is expected in spring 1985.

3. OPERATIONAL EXPERIENCE WITH CASK STORAGE

Full scale demonstration programs of complete irradiated fuel assemblies being stored in different types of GNS and TN shipping/storage casks are in progress in the FRG since 1982 (cf. Fig. 1). The objectives of these demonstrations are to verify cask design parameters, to gain operational experience in cask handling and to expand the data base on dry storage fuel performance.
The irradiated fuel assemblies were loaded into the casks in the storage pools of the German reactors, e.g. Würgassen, Stade and Biblis. The cask drying operations, which are required to reduce cover gas moisture, were carried out successfully. Fuel is stored in Helium gas at <1 atm pressure. The peak cladding temperatures decreased rapidly from about 400°C to about 200°C over the 2-years storage period, showing a substantial longitudinal and axial temperature gradient. The radiation levels are in the expected range.

The program results could be summarized as follows:

- In-pool loading and unloading of shipping/storage casks have been successfully demonstrated.
- Cask design parameters have been verified in practice.
- Radiation levels to operators are extremely low.
No rods have failed due to dry storage even for a wide temperature range and range of cladding metallurgical conditions.

4. FUEL CLADDING INTEGRITY IN DRY STORAGE

Even under the hypothetical assumption of a 100% rod failure the German cask concept and its barrier system constitutes a safe containment. However with regard to the fuel unloading procedure the question of fuel integrity and fission gas release fraction might be of interest, e.g. for operation of a fuel receiving station in a reprocessing plant.

Full scale dry storage demonstrations with different types of casks, hot cell tests with failed and unfailed fuel rods, and fuel behaviour modelling have been conducted in the FRG, using PWR and BWR Zircaloy clad fuel (cf. Fig. 2).

<table>
<thead>
<tr>
<th>Type of Storage</th>
<th>Type of Fuel</th>
<th>Total Number of Fuel Rods</th>
<th>Av. Burnup Gwd/MTU</th>
<th>Starting Temperature °C</th>
<th>Cooler Gas</th>
<th>Operating Time, Month 1 July 1984</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hot Cell</td>
<td>PWR</td>
<td>9</td>
<td>33</td>
<td>const. 400/450</td>
<td>Helium</td>
<td>18</td>
</tr>
<tr>
<td>Hot Cell</td>
<td>BWR</td>
<td>14</td>
<td>24</td>
<td>const. 400/450</td>
<td>Helium</td>
<td>18</td>
</tr>
<tr>
<td>Test Module</td>
<td>PWR</td>
<td>180</td>
<td>&lt; 43.7</td>
<td>&lt; 300</td>
<td>Moist N₂</td>
<td>3</td>
</tr>
<tr>
<td>Test Module</td>
<td>PWR</td>
<td>180</td>
<td>&lt; 41.7</td>
<td>&lt; 400</td>
<td>Moist N₂</td>
<td>12</td>
</tr>
<tr>
<td>Cask</td>
<td>BWR</td>
<td>1760</td>
<td>34.5</td>
<td>&lt; 430</td>
<td>Helium</td>
<td>17</td>
</tr>
<tr>
<td>Cask</td>
<td>PWR</td>
<td>1760</td>
<td>34.5</td>
<td>&lt; 430</td>
<td>Helium</td>
<td>17</td>
</tr>
</tbody>
</table>

FIGURE 2 Spent Fuel Performance Testing Matrix
Irradiated fuel has been placed in and retrieved from various CASTOR casks and from a TW 1300 cask.

Dry storage tests and demonstrations have involved ~3,000 fuel rods, which have been monitored during dry storage with maximum cladding temperatures ranging from 250 to 450°C. Although some tests are still in progress, there is no evidence that any rods exposed to inert gases have failed during dry storage. Even under conditions more severe than in the casks, the fuel shows no cladding failures.

Dry storage under cask conditions with maximum insertion temperatures in the range of 400°C does not expect to cause

- a significant creep deformation above 1%
- cladding oxidation
- conditions that induce SCC or flaw propagation

Thus peak cladding temperatures up to about 450°C (depending on the character of the fuel) are predicted to be acceptable for inert gas storage.
Ontario Hydro will need additional storage facilities in the early 1990's. After cooling the irradiated fuel in station waterpools, it may be advantageous to use dry storage facilities instead of more waterpools. To evaluate the potential technical and cost benefits of dry storage facilities, the conceptual development of concrete canister and dry vault storage systems is being undertaken. In addressing the dry storage systems, a generic approach has been followed.

Two alternative dry vault designs were proposed. In one design, the storage tubes are rectangular in shape to accommodate existing rectangular waterpool fuel racks, while the other has cylindrical tubes and uses high storage density cylindrical containers. The latter could potentially also be used in the fuel transportation and disposal phases, thereby reducing fuel handling. The results indicate that the dry vault design using the existing waterpool racks (trays and modules) is preferable for the storage phase alone.

Ontario Hydro is also undertaking a development program to assess the feasibility of an integrated concrete cask system to meet the requirements of storage, transportation and disposal of fuel. Progress to date has concentrated in the development of high compressive strength (=90 MPa) and low pH (=10) cement pastes and concrete mixes. Also, half scale model tests are being tested to evaluate their impact and fire resistance during transportation accidents and their thermal behaviour during storage and disposal.
1. INTRODUCTION

Current international interest in the dry storage of irradiated fuel reflects the expectation that this mode of storage will reduce costs over waterpool storage. This mainly results from using passive and modular systems that require low capital investment and spread costs uniformly over the lifetime of the facility by allowing the building of facilities as the need for additional storage arises. Modularity provides flexibility and reduces the financial burdens, while passivity eliminates the need for maintaining continuous operating equipment.

The advantages of dry storage over waterpool storage led to the creation of demonstration programs in several countries, and most notably in the U.K. where this led to the commercial application of the dry vault concept at Wylfa. At Ontario Hydro, a program to evaluate dry storage has been on-going since 1980. This program was initiated following the successful storage of irradiated fuel in concrete canisters at the AECL Whiteshell Nuclear Research Establishment. In 1980, Ontario Hydro joined AECL in an experimental program to study the long-term behaviour of irradiated CANDU fuel when stored in dry and moisture-saturated air at 150°C in concrete canisters (K. Wasywich et al. 1984). At the same time, design studies were begun to evaluate further the technical and economic benefits of dry storage. Currently, several options are being considered: (1) concrete canisters, (2) dry vaults, (3) concrete integrated casks (4) cast-iron casks. However, concept work with the cast iron cask has just begun and thus the first three options only are considered in the rest of this report. The options are being developed to (i) meet the specific needs of Ontario Hydro's nuclear generation stations, and (ii) a stage such that commercial scale storage facilities can be built. Emphasis is being given to site-specific features of these concepts. Following this development phase, studies of a prototype nature will be conducted to gain hands-on
experience and support their implementation as the next additional storage facilities.

The preliminary phase of the dry storage program has centered on the generic evaluation of these concepts. The results from these generic studies are the main focus of this paper.

2. CONCRETE CANISTER SYSTEM

Concrete canisters are low cost modular units built as needed. The canisters are passive with the decay heat of the fuel dissipated by free convection. The generic system proposed (J. Freire-Canosa, 1983a) consists of four main components:

1. concrete canisters;

2. fuel handling facility;

3. storage yard for the canisters;

4. handling equipment.

Handling of the irradiated fuel within this system follows four stages:

1. Fuel is transferred in the waterpool modules/trays to the fuel handling facility with an on-site metallic cask.

2. At the fuel handling facility, fuel is transferred from the modules/trays to a metallic container and seal-welded.
3. The metallic containers (four in total) are loaded into the concrete canister.

4. The concrete canister is transported to the storage yard.

On retrieving fuel from storage, the procedure is reversed. It must be emphasized that the approach followed in this preliminary evaluation was to have a comparison with the waterpool that will reflect a conservative design in the requirements for the dry storage systems. Hence, in the generic development of a concrete canister system, the building of a hot-cell facility to transfer fuel from the waterpool storage racks to a high storage density container was considered. Currently, our design efforts are directed towards eliminating this hot-cell facility by means of a fuel transfer system that would allow full use of already available facilities on the site and reduce costs even further.

2.1 THE CONCRETE CANISTER

The concrete canisters being proposed are monoliths of reinforced ordinary concrete with an inner sealed welded steel lined cavity to form a closed system to the environment. The canisters, which have a capacity of 800 CANDU bundles (17 MgU), weigh 150 Mg when loaded. The canisters are designed to store 10 year old fuel and their main physical characteristics are summarized in Table 1 and illustrated in Figure 1. Fuel is located within the canister in seal-welded carbon steel containers designed for high storage density and fuel storage temperatures below 130°C. The bundles are positioned within a square lattice grid designed to allow free convectivity. After the first few years of production, the total concrete canister cost is estimated at 59k$ in 1983 dollars, including the cost for four metallic containers.
<table>
<thead>
<tr>
<th>Characteristic</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Capacity (No. of Bundles)</td>
<td>800</td>
</tr>
<tr>
<td>Storage Configuration</td>
<td>Square</td>
</tr>
<tr>
<td>Approximate Payload (empty)</td>
<td>125 Mg</td>
</tr>
<tr>
<td>Approximate Payload (full)</td>
<td>150 Mg</td>
</tr>
<tr>
<td>Height</td>
<td>6.21 m</td>
</tr>
<tr>
<td>Overall Cross-section</td>
<td>2.98 m x 2.98 m</td>
</tr>
<tr>
<td>Inner Cross-section</td>
<td>1.12 m</td>
</tr>
<tr>
<td>Thickness Concrete Shell</td>
<td>0.91 m</td>
</tr>
<tr>
<td>Thickness Steel Liner</td>
<td>0.015 m</td>
</tr>
<tr>
<td>Concrete Type</td>
<td>Ordinary</td>
</tr>
<tr>
<td>Concrete Density (Mg/m³)</td>
<td>2.3</td>
</tr>
</tbody>
</table>
**Figure 1. SQUARE CONCRETE CANISTER**

- **Bundle Storage Capacity:** 600 bundles in 4 contains
- **Weight (Empty):** 125 Mg
- **Weight (Loaded):** 150 Mg

**Dimensions:**
- **Inner Liner:**
- **Storage Cavity:**
- **Concrete Shell:**
- **Capital Cost:** ~35 k$
2.2 FUEL HANDLING FACILITY

The fuel handling facility acts as the interface between the waterpool and the concrete canisters. Its main functional requirement is the transferring of fuel from the waterpool storage racks to the concrete canisters. The total cost for this facility is estimated at 10M$ (1983 dollars).

The proposed facility is shown in Figure 2. It is subdivided into three main areas: office and change room area, hot-cell and ancillary services. The hot-cell is located 11 m above ground level to allow the canister and on/off site transportation casks to be positioned in the two corridors underneath its 1 m thick floor. Canister and cask will be located under the two port-holes that open directly into the hot-cell. The fuel is transferred into and out of the hot-cell through these port-holes. While loading/removing fuel into the hot-cell, shielding sleeves are lowered onto the canister and transport cask to shield personnel from radiation while working in the corridors. Decontamination capability is also provided in the corridors.

At the hot-cell the fuel is transferred from the waterpool modules/trays to the square grid containers with a transfer machine. Once full, they are seal welded and leak tested. The containers are then lowered by the hot-cell overhead crane through a port-hole into the concrete canister below. When the canister is loaded, its shielding plug is put in place and seal-welded. At this point, the canister is ready for storage.

2.3 STORAGE YARD

The storage yard is a reserved fenced area within the site for storing the concrete canisters with the fuel. The ground is paved gravel with loading capacity of 80-100 Mg/m². The canisters are arranged into panels with a 7.5 m road in between. Within the
LEGEND:
1. Change Area and Offices Wing
2. Hot-Cell
3. Service Wing
4. Hot-Bay for Handling On-Site Cask
5. Hot-Bay for Handling Concrete Canister
6. Hot-Cell Corridor
7. Port-Holes for Hot-Bays
8. Port-Holes for Hot-Bays
9. Hot-Cell Access Man-Door

Capital Cost ≈ 10 M $
panels, the canisters are laid horizontally in rows as shown in Figure 3. Stacks of canisters in a cross-wise fashion is being proposed with a 0.5 m spacing between rows. The dose at the surface of the canisters is about 2 μSv/h and will not exceed 8 μSv/h anywhere within the facility.

2.4 EQUIPMENT

Equipment is mainly needed to transport and handle the concrete canisters and do fuel transfers. A tractor and multiple axle trailer (5 m wide x 11 m long) capable of moving 250 Mg is proposed to transport the canisters on the site. A crane with 300 Mg capacity will be dedicated at the storage yard. At the hot-cell, a fuel transfer machine, a leak detection system, an automatic welding machine and an overhead crane are the main equipment requirements.

3. DRY VAULT SYSTEM

Several generic dry vault concepts based on passive cooling and multiple barriers to minimize the potential for release of radioactive materials are being studied at Ontario Hydro (J. Cipolla, 1983a and 1984b)). A view of the dry vault structure showing a cross-section of a typical storage cell is illustrated in Figure 4. The dry vault structure consists of three main areas:

1. Fuel handling and services;
2. Storage cells;
3. Transfer hall.
LEGEND:
1. Fuel Handling Facility
2. Storage Yard
3. Storage Yard Operations
4. Concrete Canister Arrays

Figure 3. PANORAMIC VIEW OF THE CONCRETE CANISTER SYSTEM
The structure covers an area of 2500 m$^2$ and is divided in four storage cells with capacity for 350,000 (= 7400 MgU) CANDU fuel bundles. The fuel handling and services area houses the fuel reception area, control room for the facility, hot-cell, decontamination rooms, equipment and storage areas. Fuel handling is performed within the hot-cell where drying, fuel transfer and containerization equipment is located.

Irradiated Fuel stored at the waterpools is transported first to the hot-cell in their waterpool storage racks by an on-site metallic cask. The fuel is then transferred to a metallic container, seal-welded and removed from the hot-cell by the dry vault fuel transfer mechanism and emplaced in the storage tubes. This fuel transfer mechanism is a lead-shielded shroud fixed to an overhead crane with a hoist and grappling device to handle the metallic containers.

The storage cells are situated on either side of the fuel handling and services area. Irradiated fuel is stored within carbon-steel tube arrays, opening to the dry vault hall. Concrete plugs located at the tube openings provide shielding from the radiation of the stored fuel. Containment is provided by three barriers:

1. Seal-welded carbon steel containers enclosing the fuel;
2. the storage cell tubes;
3. emergency air exhaust system.

A typical storage cell can hold up to 90,000 bundles (= 1900 MgU) in 15 m long tubes and is modular in design. The temperature of the fuel is not expected to exceed 165°C.
Figure 4. Dry Vault Facility
General View and Cross-Section
of a Typical Storage Cell
The transfer hall spans the length of the dry vault and is located above the storage cells and fuel handling and service area. The hall provides access to the hot-cell and storage cells and serves to transfer fuel from the hot-cell to the storage tubes and vice versa. Its superstructure is a conventional metallic shell.

3.1 DRY VAULT STORAGE ALTERNATIVES

Two alternate dry vaults were evaluated:

1. Option 1: the dry vault stores fuel in the waterpool storage racks (modules or trays) to eliminate an extra handling step. The storage racks are enclosed within seal-welded rectangular containers (384 bundles/container) and then stored in rectangular tubes at the storage cell (Figure 5).

2. Option 2: The dry vault stores fuel in cylindrical containers of high storage density (288 bundles/container). Transfer of fuel from the waterpool racks to these containers is done in the hot-cell. These containers are also relevant to the integration of irradiated fuel management because with minor modifications they can be designed for use in the transportation and disposal phases (Figure 5).

In Option 1, the fuel is stored in rectangular tubes arranged in a square pitch. In Option 2, the fuel is in cylindrical tubes arranged in a triangular pitch. A summary of the dry vault cell capacities is given in Table 3. Use of rectangular or cylindrical tubes does not alter the generic conceptual design philosophy of the dry vault examined at Ontario Hydro.
TABLE 2

PHYSICAL CHARACTERISTICS OF DRY VAULT STORAGE CELLS

<table>
<thead>
<tr>
<th>Option 1: Storage of</th>
<th>BRUCE</th>
<th>PICKERING AND DARLINGTON</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>No. of Tubes/ Cell</td>
<td>Bundle/ Tube</td>
</tr>
<tr>
<td>Waterpool racks</td>
<td>36</td>
<td>2280</td>
</tr>
<tr>
<td>Option 2: High density cylindrical containers</td>
<td>77</td>
<td>1152</td>
</tr>
</tbody>
</table>

4. CONCRETE INTEGRATED CASK PROGRAM

Following a technical and economic evaluation on the use of concrete casks as universal containers for storage, transportation and disposal of irradiated CANDU fuel (J. Freire-Canosa, 1982b and c, N.C. Burnett, 1982 and 1984) a development program was established to assess the feasibility of this scheme. Design criteria, based on current international regulations but conservative to allow for future possible changes in these regulations, were specified for the development of new concretes and the construction of half scale model casks.

An irradiated fuel management system which uses an integrated cask has the potential over a multicontainer system to:

1. Reduce capital and operating costs by utilizing modular storage;
Option 1
- Holds 4 Modules
- Stores 384 Bundles

Option 2
- Holds 48 Cartridge Tubes
- Stores 288 Bundles

Figure 5. Dry Vault Storage: Container Options
2. Increase operational flexibility which allows for emergency on long term storage at any point during management of irradiated fuel.

3. Decrease the number of fuel handling operations.

The development program is divided into three main areas:

<table>
<thead>
<tr>
<th>Area</th>
<th>Objective</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Concrete Technology Development</td>
<td>To produce a high strength, high density, low permeability concrete</td>
</tr>
<tr>
<td>2. Concrete Durability</td>
<td>To test the ability of the concrete to be an effective and inert barrier in a repository</td>
</tr>
<tr>
<td>3. Cask Development:</td>
<td>To design and fabricate a prototype cask. Testing of the cask to a number of the performance requirements.</td>
</tr>
<tr>
<td>Design and Testing</td>
<td></td>
</tr>
</tbody>
</table>

4.1.1 **CONCRETE TECHNOLOGY DEVELOPMENT**

Research in this area has focused on formulating and testing a number of mix compositions to obtain the required properties. The following materials have been included:

1. A sulphate resisting portland cement conforming to ASTM Type V, CSA Type 50 and API Class G requirements.
2. ASTM Class F flyash or a condensed silica fume (supplementary cementing materials).


4. Magnetite/specularite aggregates.

Compositions were fabricated at varying water-solids ratios and samples cured at room temperature. Curing of samples at elevated temperatures is planned for the future.

A summary of the design guidelines and experimental results is shown in Table 3. It should be noted here that an increase in water-solids ratio from 0.25 to 0.36 will result in increased workability, but decrease the compressive strength by 25-30 percent and also increase porosity and permeability. Hence it is important when considering cask fabrication, to concentrate on additives and emplacement methods which will keep the water-solids ratio as low as possible.

4.2 CONCRETE DURABILITY

For fuel disposal, it must be demonstrated that the cask has the ability to last for several hundred years in a potentially hostile environment. As such, a number of experiments are planned to study:

1. the reactions in the system: bentonite clay-concrete-used fuel-groundwater;

2. the effect of concrete on groundwater chemistry and the subsequent effect on $\text{UO}_2$ dissolution;

3. the radionuclide retaining capability of concrete;
<table>
<thead>
<tr>
<th>Parameter</th>
<th>Requirement</th>
<th>Rationale</th>
<th>Optimum Results</th>
</tr>
</thead>
<tbody>
<tr>
<td>Density (ρ)</td>
<td>3.5 Mg/m³</td>
<td>The density of Ordinary Portland Cement concrete is 2.4 Mg/m³. Increasing this density by using iron-based aggregates allows the cask walls to be thinner, yet maintaining the radiation shielding capability.</td>
<td>ρ = 3.65 Mg/m³</td>
</tr>
<tr>
<td>Thermal Conductivity (K)</td>
<td>2.5 W/m°C</td>
<td>The thermal conductivity of ordinary concrete is 1.6 W/m°C. Introduction of an iron-based aggregate increases this substantially, resulting in higher heat dissipation and lower internal cask and fuel temperatures.</td>
<td>Not available</td>
</tr>
</tbody>
</table>
**TABLE 3 (Cont'd)**

**CONCRETE DEVELOPMENT**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Requirement</th>
<th>Rationale</th>
<th>Optimum Results</th>
</tr>
</thead>
<tbody>
<tr>
<td>Compressive Strength (fc)</td>
<td>50 MPa</td>
<td>Structural strength of the cask is dependent on the rebar design and the concrete itself. Improving the compressive strength of the concrete will help the cask withstand impacts, resulting from transportation accidents, and potentially high hydrostatic pressure resulting from disposal vault flooding.</td>
<td>These results are from analysis of the optimum cement paste, not a concrete W/S = 0.25</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Time After Hydration 28 Days 91 Days 182 Days</td>
<td>fc (MPa)</td>
</tr>
<tr>
<td>Permeability (K)</td>
<td>$10^{-13}$ ms$^{-1}$</td>
<td>The concrete cask walls should have minimum permeability to protect the fuel/waste against groundwater intrusion into the disposal vault.</td>
<td>K($10^{-13}$ ms$^{-1}$) Total Porosity (%)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>9.8</td>
</tr>
<tr>
<td>Alkalinity (pH)</td>
<td>11 (maximum)</td>
<td>If the alkalinity of the groundwater increases as it eventually migrates through the concrete, the subsequent dissolution rate of the fuel/waste may increase.</td>
<td>Ca(OH)$_2$ (%) pH</td>
</tr>
</tbody>
</table>
4. the radiation stability of concrete.

Experiments in (1) are now underway at Atomic Energy of Canada Ltd. Initial results show that although the alkalinity of deionized water containing concrete raises sharply to pH 11, that of Canadian shield groundwater containing bentonite clay and concrete does not rise above pH 7.

4.3 CASK DEVELOPMENT: DESIGN AND TESTING

This phase of the program involves optimization of cask shape, dimensions, weight and fuel payload. Shielding calculations were based on 10 year cooled CANDU fuel and the prototype design studies included cask reinforcement, lid sealing and cask lifting arrangements. Based on the preliminary analysis of

1. the ability to withstand certain mechanical and thermal loads;

2. storage efficiency and cost;

3. handling, maneuverability and constructability.

The following initial conceptual design was adopted (see Figure 6):

Shape - cylindrical

Payload - 3.5 Mg of 10 year cooled CANDU fuel

Cask Weight - 42 Mg

Cask Size - 2.75 m (dia) x 2.41 m (high)
Wall Thickness  -  0.53 m high density concrete

Construction  -  reinforced concrete with inner steel liner

Increasing the payload to 5.2 Mg increases the cask weight to 54 Mg and cask height to 3.0 m. The inner steel liner is considered as the primary containment barrier and the concrete as the 'universal overpack', acting as radiation shield, impact and fire protection and as a barrier to water ingress and radionuclide migration in the repository. From initial analysis, the temperature gradient through the cask wall is a maximum of 20°C, under a variety of environmental conditions, for a heat loading of 1.3 kW. It is projected that with this design, a loading at least 5 kW is possible without affecting the structural or shielding integrity of the wall.

The main thrust of this work at the present time is the construction of two half-scale models based on the conceptual design. These will be used to measure:

1. thermal stresses and thermal gradients within the cask under a variety of heat loadings, environmental conditions and power cycles. Crack initiation and propagation will be examined in detail;

2. cask performance under simulated transport accident conditions according to the IAEA guidelines (drop tests and fire test). This work is being performed at the Chalk River Nuclear Laboratories.

A detailed economic evaluation of the integrated system has not yet been undertaken, only a preliminary study (J. Freire-Canosa, 1982 a and b) which showed its potential benefits.
4.4 FUTURE CONSIDERATIONS

The results of this preliminary development program will shortly be assessed to determine the viability of using concrete casks for an integrated system. A 'universal' cask designed for this type of system may not be the most economically attractive option for a particular phase such as storage. However, when undertaking an economic assessment to determine the cost advantages of the integrated system, the overall irradiated fuel management strategy must be considered. The operational benefits of reducing fuel handling steps and thus occupational radiation dose uptake make this option extremely attractive.

From this study, an option which may show promise, both economically and operationally, involves a cask to meet the storage and disposal criteria, but with the added protection required for transportation, provided by a fully licensed overpack.

5. CONCLUSIONS

The concrete canister and dry vault concepts for the storage of irradiated CANDU fuel were developed and assessed. This study examined the incentive to provide additional storage capacity in place of waterpools for the Ontario Hydro committed nuclear program. A further (1985) site specific evaluation of the alterations will assess the economics in detail:
Figure 6. PROTOTYPE MODEL FOR A CONCRETE INTEGRATED CASK

Figure 6 shows a diagram of a prototype model for a concrete integrated cask. The diagram includes the following components:

- **Cask Lifting Lugs**
- **Lid Lifting Lugs**
- **Seal Grout**
- **Two Rectangular Storage and Shipping Fuel Modules (192 Fuel Bundles)**
- **Steel Liner**
- **Elastomer Seal**
- **Reinforcing Bars**
- **Heavy Shielding Concrete** designed for durability, impact and fire resistance

**Approximate Size**
- Diameter: 2.75 m
- Height: 2.41 m
- Weight: 45 - 47 Te
REFERENCES


A metallic cask for the dry storage of LWR irradiated fuel is being developed in Spain. In 1981 three Spanish organizations, Empresa Nacional del Urano, S.A. (ENUSA), Junta de Energia Nuclear (JEN) and Equipos Nucleares, S.A. (ENSA) initiated a joint program to achieve the design and the fabrication of a dry storage cask prototype.

The main design parameters are: capacity for 17 PWR spent fuel assemblies with 1 KW/assembly residual thermal power. The maximum weight shall be 115 metric tonnes and it will be licensed for transport outside of nuclear power sites.
1. INTRODUCTION

The Spanish nuclear power program has been revised this year. The National Energy Plan (PEN), recently approved by the Parliament, includes 10 nuclear power stations as shown in Table 1. All the nuclear power stations are of LWR type, excepting Vandellos I (gas cooled, natural uranium reactor). The spent fuel from this reactor (80–90 tU/year) is being reprocessed by COGEMA, France.

The total LWR spent fuel arisings are estimated at about 5300 tonnes U. Of this spent fuel, 114 tonnes, from Zorita (56 tU) and Garona (88 tU), were delivered to British Nuclear Fuel, Ltd., UK.

According to this program, short and medium term plans have been established to assure the spent fuel management. At short term, utilities have increased the storage capacity of spent fuel pools by installing compact racks. The storage capacity of nuclear reactors is shown on Table 2. At medium and longer term, dry storage of spent fuel is planned.

For this reason, the design and construction of a prototype dry storage metallic cask, similar to the spent fuel shipping casks, was initiated in 1981. The present situation of the different stages, design and fabrication are outlined in this paper.

2. TECHNICAL SPECIFICATIONS

The main objective of this project is to develop the design and fabrication of a metallic cask prototype for the dry storage of light water reactor spent fuel, able to be shipped outside nuclear power station sites, according with international regulations and competitive in the international market with other casks for the same purpose.
Table 3 shows the main technical specifications established for the design. The cask shall comply with the requirements of the fissile class II, Type B (U) containers, as defined in the IAEA regulations, as well as the internationally accepted standards and regulations for storage of spent nuclear fuel.

The maximum weight is not to exceed 115 tonnes fully loaded. The design contemplates wet loading in the reactor pool. Maximum length and diameter shall be 500 cm and 250 cm respectively.

The first prototype shall be designed to contain 17 fuel assemblies (standard 17 x 17 PWR type Westinghouse design). Calculations of radioactive and neutronic characteristics of the spent fuel are based on 3.5% U-235 initial enrichment and 40,000 MWD/tU maximum burn-up.

3. DESIGN DEVELOPMENT

The main stages of the design are the following:

- conceptual design
- validation of analytical codes
- radioactive and neutronic characteristics of spent fuel
- development of the cask fabrication materials
- thermal design, including heat transfer tests
- shielding design
- drop test evaluation

The conceptual design covers among others, the following items:

- nodular cast iron as containment vessel material
- aluminum or aluminum alloys as basket material
- boral or borated stainless steel as thermal neutron absorber
- borosilicone as fast neutron shielding
- 250°C maximum temperature of fuel rod cladding
- 150 cm internal diameter
- air or inert gas atmosphere
- two lids to assure double barrier containment

3.1 RADIATION SOURCES

The radiation sources have been calculated using the ORIGEN-2 computer code. For a decay heat of 1 Kw per assembly, the cooling time is 5 years, corresponding to a gamma source of $8.85 \times 10^{15}$ photons/sec/assembly and a neutron source of $2.17 \times 10^8$ neutrons/sec/assembly.

3.2 NUCLEAR DESIGN

The nuclear criticality analysis has been made in two ways, using KENO IV and TWOTRAN-TRACA computer codes. The last one is a version of the TWOTRAN-GG Code, modified by the Junta de Energia Nuclear (JEN).

A subcritical design ($K_{eff} < 0.95$) of the basket has been selected, using borated stainless steel (1.3% boron) as neutron absorber.

3.3 THERMAL DESIGN

The design heat sources are the decay heat from spent fuel assemblies (17 KW) and solar radiation. The cask must be designed to withstand the fire test conditions as determined by IAEA (800°C during 30 minutes). The design temperatures are: exterior cask surface less than 82°C, considering an environmental temperature of 38°C, and cladding temperature of fuel rods less than 250°C.

An experimental thermal rig has been developed in order to study axial and radial heat transfer from the spent fuel to the basket. The experiment contains a full scale simulated fuel element (PWR 17 x 17)
with electrically heated tubes housed in an isolated aluminum box. The
heat generation rate is adjusted by the voltage applied to the
sub-bundles and their measured electrical resistance. The temperatures
are recorded by chromel-alumel thermocouples distributed on the inside
and outside of the tubes. Two different atmospheres, air and helium, are
being studied. Some experimental values are given in Table 4.

The radial temperature distribution is being calculated with the
aid of the heat transfer computer code HEATING-5.

3.4 SHIELDING DESIGN

Shielding design of the dry storage cask will involve the use of
three shielding computer codes, SABINE-3, ANISN and MORSE. At present,
shielding evaluation has been only made using SABINE-3, external surface
doses resulting less than 40 millirem/hour.

4. FABRICATION DEVELOPMENT

In accordance with IAEA regulations, the material of the
containment has been selected on the basis of its fracture toughness
which requires ductile behaviour at low temperatures (minus 40°C). A
nodular iron casting has been developed by Babcock and Wilcox Espanola
(BWE) according with the specifications of Equipos Nucleares (ENSA), who
is developing the mechanical design and fabrication procedures. At
present, a reduced scale (1:2) prototype has been cast by BWE.

Two computer codes, DYNA 3D and HONDO, are being used for the
prediction of the 9 meter drop test behaviour and the design of the
sealing surfaces of the lids.
5. FUTURE DEVELOPMENT

The developed activities can be summarized as follows:

- conceptual design
- computer codes for analytical studies
- first series of thermal test runs
- design of scale model prototype (1:2)
- casting of the body of a scale model prototype (1:2)

The main activities still to be developed are as follows:

- construction of a test facility
- testing of the scale prototype
- second series of thermal experiment runs
- detailed basket design and fabrication
- casting of other scale prototypes, including a full scale model
### Table 1

**Spanish Nuclear Program**

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Type</th>
<th>Power, MWe</th>
<th>Start Up, Year</th>
</tr>
</thead>
<tbody>
<tr>
<td>Zorita</td>
<td>PWR</td>
<td>160</td>
<td>1968</td>
</tr>
<tr>
<td>Garona</td>
<td>BWR</td>
<td>460</td>
<td>1971</td>
</tr>
<tr>
<td>Vandellos I</td>
<td>GCR</td>
<td>480</td>
<td>1972</td>
</tr>
<tr>
<td>Almaraz I</td>
<td>PWR</td>
<td>930</td>
<td>1981</td>
</tr>
<tr>
<td>Asco</td>
<td>PWR</td>
<td>930</td>
<td>1983</td>
</tr>
<tr>
<td>Almaraz II</td>
<td>PWR</td>
<td>930</td>
<td>1983</td>
</tr>
<tr>
<td>Cofrentes</td>
<td>BWR</td>
<td>975</td>
<td>1984</td>
</tr>
<tr>
<td>Asco II</td>
<td>PWR</td>
<td>930</td>
<td>1984</td>
</tr>
<tr>
<td>Trillo I</td>
<td>PWR</td>
<td>~1000</td>
<td>1988</td>
</tr>
<tr>
<td>Vandellos II</td>
<td>PWR</td>
<td>~1000</td>
<td>1990</td>
</tr>
</tbody>
</table>
**TABLE 2**

**REACTOR STORAGE CAPACITY**

<table>
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<tr>
<th>Reactor</th>
<th>Core (Fuel Assemblies)</th>
<th>Pool (Fuel Assemblies)</th>
<th>Effective</th>
<th>Saturation (Year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Zorita</td>
<td>69</td>
<td>310</td>
<td>241</td>
<td>1995-1997</td>
</tr>
<tr>
<td>Garona</td>
<td>400</td>
<td>1727</td>
<td>1327</td>
<td>1997-1998</td>
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<tr>
<td>Almaraz I</td>
<td>157</td>
<td>622</td>
<td>465</td>
<td>1994-1996</td>
</tr>
<tr>
<td>Asco I</td>
<td>157</td>
<td>588</td>
<td>431</td>
<td>1994-1996</td>
</tr>
<tr>
<td>Almaraz II</td>
<td>157</td>
<td>622</td>
<td>465</td>
<td>1995-1997</td>
</tr>
<tr>
<td>Cofrentes</td>
<td>624</td>
<td>3088</td>
<td>2464</td>
<td>2003-2006</td>
</tr>
<tr>
<td>Asco II</td>
<td>157</td>
<td>588</td>
<td>431</td>
<td>1995-1997</td>
</tr>
<tr>
<td>Trillo I</td>
<td>177</td>
<td>600</td>
<td>423</td>
<td>1995-1997</td>
</tr>
<tr>
<td>Vandellos II</td>
<td>157</td>
<td>560</td>
<td>403</td>
<td>1997-1999</td>
</tr>
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TABLE 3
TECHNICAL SPECIFICATIONS

<table>
<thead>
<tr>
<th>Specification</th>
<th>Details</th>
</tr>
</thead>
<tbody>
<tr>
<td>Capacity</td>
<td>17 PWR fuel assemblies (7.8 tU)</td>
</tr>
<tr>
<td>Thermal Rating</td>
<td>17 Kw</td>
</tr>
<tr>
<td>Initial Enrichment</td>
<td>3.5% U-235</td>
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<tr>
<td>Maximum Burnup</td>
<td>40,000 MWD/tU</td>
</tr>
<tr>
<td>Maximum Total Weight</td>
<td>115 tonnes</td>
</tr>
<tr>
<td>Maximum Height</td>
<td>5,000 mm</td>
</tr>
<tr>
<td>Maximum External Diameter</td>
<td>2,500 mm</td>
</tr>
<tr>
<td>Material Cask</td>
<td>Nodular Cask Iron</td>
</tr>
<tr>
<td>Material Basket</td>
<td>Aluminum or Aluminum alloy</td>
</tr>
<tr>
<td>Standards and Regulations</td>
<td>IAEA Recommendations</td>
</tr>
<tr>
<td></td>
<td>10 CFR 72 (USA-NRC)</td>
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<tr>
<td>Fuel Assembly</td>
<td>Standard 17 x 17 PWR Westinghouse Design</td>
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<tr>
<td></td>
<td>Air</td>
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<tr>
<td>------------</td>
<td>------</td>
</tr>
<tr>
<td>Max. Fuel Rod Temperature (°C)</td>
<td>143</td>
</tr>
<tr>
<td></td>
<td>156</td>
</tr>
<tr>
<td>Max. Aluminum Box Temperature (°C)</td>
<td>190</td>
</tr>
<tr>
<td></td>
<td>200</td>
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<td>Power Level (W)</td>
<td>223</td>
</tr>
<tr>
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<td>134</td>
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<td>168</td>
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<tr>
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<td>201</td>
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<td>209</td>
</tr>
</tbody>
</table>
DRY STORAGE FACILITY FOR SPENT FUELS IN JAPAN*

M. Adachi, K. Sakano, K. Ishimoto, A. Yamamoto
Japan Atomic Energy Research Institute
Tokai-Mura, Ibaraki-Ken, Japan

ABSTRACT

To store the JRR-3 metallic natural uranium spent fuel elements, a dry storage facility has been constructed in JAERI. This facility has a capacity of about 30 tons of uranium. The store is basically an ordinary concrete box about 12 m long, 13 m wide and 5 m deep. The store comprises a 10 x 10 lattice array of the drywells. The drywell has an air-inlet and -outlet pipe for radiation monitoring and a carbon steel shield plug for radiation protection.

A canister, which is made of stainless steel 0.5 cm thick, can accommodate 36 fuel elements. The canister is sealed by fusion welding.

The results of the safety assessment indicate that the environmental impact under normal operation and possible accidents in the dry storage facility is negligibly small.

The transportation of spent fuel elements and the operation of the facility began in November 1982. About 1,800 fuel elements (15 tons U) are stored in the facility at present.

We have experience with the dry storage of spent fuel elements for 2 years. It is found that there is no measurable increase of radioactivity in the air circulatory systems.

*Paper submitted too late for presentation at the Workshop.
1. INTRODUCTION

To store the JRR-3 metallic natural uranium spent fuel elements, a dry storage facility has been constructed in JAERI. This facility has a capacity of about 30 tons of uranium (3600 fuel elements). This report describes the design and construction of this facility, the transportation and encapsulation of the fuel elements, and our operational experience.

Japan Research Reactor No. 3 (JRR-3) had been a heavy water moderated/cooled tank type reactor of 10 MW thermal power using metallic natural uranium fuel (aluminum cladding), which is now being modified into a light water moderated/cooled pool type reactor of 20 MW thermal power. The existing reactor has been operated for 20 years. Consequently, a large number of spent fuel elements are stored in the JRR-3 spent fuel pond. To store these spent fuel elements for an extended period of time, a dry storage facility was constructed by JAERI in February of 1982. The general arrangement of the facility is illustrated in Figure 1. This is the first away from reactor (AFR) spent fuel dry storage facility in Japan.

2. SPENT FUEL ELEMENTS TO STORE

Figure 2 shows the JRR-3 metallic natural uranium fuel assembly and elements. A fuel assembly consists of shielding plug, cooling tube and three fuel elements. The elements are connected with pins. They are metallic uranium rods of 1 in. diameter contained in the aluminum alloy
tubes with total length of about 950 mm. The fuel assemblies are cut out into the elements in the pond. The specification of an element is as follows:

<table>
<thead>
<tr>
<th>Specification</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Burn-up (max)</td>
<td>800 MWD/T</td>
</tr>
<tr>
<td>Cooling Time (max)</td>
<td>2500 days</td>
</tr>
<tr>
<td>Total FP activities (max)</td>
<td>110 Ci</td>
</tr>
<tr>
<td>Decay heat (max)</td>
<td>0.5 W</td>
</tr>
</tbody>
</table>

3. DRY STORAGE FACILITY AND EQUIPMENT

The facility consists of a storage building and an auxiliary systems room. In the storage building, there are a cask receiving area, a loading area, a cask maintenance area and a control room. The store is under the loading area, and basically is an ordinary concrete box about 12 m long, 13 m wide and 5 m deep. The spent fuel elements are contained in the encapsulated canister, and then stored in the drywell.

As illustrated in Figure 3, the drywell is composed of a stainless steel liner which is 2.5 m deep, 36 cm inner diameter and 0.8 cm thick. A drywell has an air-inlet and -outlet pipe for radiation monitoring and a shield plug of carbon steel for radiation protection.

The canister which is made of 0.5 cm thick stainless steel can accommodate 36 fuel elements. The canister is sealed by fusion welding. The details of the canister are shown in Figure 4. Figure 5 shows the arrangement of the monitoring system. Slightly subatmospheric pressure in the drywell is chosen so that any slight leakage will be inwards. The drywell maintains the pressure between 50 mb and 120 mb below atmospheric pressure.
pressure. The environmental air in the drywell is circulated by blowers during operation for radiation monitoring by the plastic scintillation counter. The location of a leaky fuel element and/or canister can be determined by valve operation. The fuel elements don't need to be cooled because of the low burn-up and long cooling time. Therefore, a cooling system is not provided. Pressure change in this system is continuously monitored to minimize the fission product gas (Kr-85) release. The system is also provided with a dehumidifier so the humidity should be kept below 20 percent relative humidity to prevent aluminum cladding corrosion. This system has been held usually in closed circuit. For filtration of exhaust air from the system, fans and filters are provided. The shield plug is bolted to the upper plane in the drywell. The drywell is sealed by a neoprene "O" ring packing between the bottom of the shield plug and the ledge in the drywell.

4. SAFETY EVALUATION

The facility is designed to meet almost the same stringent safety requirements as applied to Japanese nuclear reactors. The structure is designed to retain its integrity in usual loading and transient operation, even in an earthquake. The fuel elements are placed in the welded canisters whose design includes a shock absorber such that an accidentally dropped canister would not release radioactive materials. The shock absorber is of the honey-comb type and placed in the bottom of the drywell. Even if a number of leaky canisters and fuel elements are found, the only isotope of importance which reaches the environment is Kr-85. However, the safety analysis shows that any release of Kr-85 is below the safety criteria limits. The maximum temperature of the spent fuel elements would not exceed 45°C during the storage period because the decay heat is removed by natural convection. The shielding wall above is 1.5 m thick, so that radiation
protection should be adequate. Air inlet and outlet tubes are arranged with dogleg turns built-in to prevent direct radiation streaming. Pre-test examination demonstrated that the corrosion of aluminum cladding is negligible and the elements do obviously not need drying. The facility and handling provisions allow retrieval and detailed inspection of selected canisters, including non-destructive examination of the fuel elements in a hot laboratory.

The results of the safety assessment indicate that the environmental impact under normal operation and possible accidents in the dry storage facility is negligibly small.

5. TRANSPORTATION AND ENCAPSULATION

The storage facility is designed to accept dry fuel elements from the dry shipping cask. The encapsulated canisters are transported from the hot laboratory to the dry storage facility by the vehicle. The details of the shipping cask are shown in Figure 6. The shipping cask is a stainless steel covered lead cylindrical structure, about 160 cm in nominal diameter and 260 cm high. The total weight is about 18 tons. The design of the cask and its handling provisions are such that an accidentally dropped cask would not release radioactive materials. The shipping cask is removed using the overhead crane provided in the dry storage facility. The crane has a capacity of 30 tons. A gamma-ray shielding gate is applied when the canister is loaded into the drywell. This gamma-ray shielding gate is box shaped and is a stainless steel covered lead structure.
The canisters are encapsulated in the hot laboratory near JRR-3. The procedure flow in the hot laboratory is shown in Figure 7. Fusion welding of the canister is carried out by a TIG welding machine in the cell. This machine is treated specifically by remote operation. The canister, shown in Figure 4, consists of the body and the closure lid. The closure lid consists of a flat disc and a handling head of stainless steel. The canister is sealed by fusion welding of a small lip of the closure lid to the upper surface of the canister body. This small lip is machined as a part of the closure lid. The air in the canister is evacuated and then backfilled with helium to atmospheric pressure through the top of the handling head. After complete helium filling, a small plug is inserted into the handling head. In this work, helium is maintained in the canister by springs and metallic packing provided in the closure lid. Then, the canister is sealed by fusion welding between the small lip of inserted plug and the top of the handling head. After the canister filling with helium, various inspections are performed. These inspections include visual inspection, a helium leak check and a surface contamination check. Visual inspections are performed with a periscope and helium leak checks are performed in the vacuum chamber in the cell. If a leaky canister is detected by the helium leak check, the canister is cut by a diamond cutter provided in the decanning machine. Then the fuel elements are taken out, and recanned in another canister. To check surface contamination, the operator takes filter paper smears on the canister surface using a manipulator. If contamination is found, the canister surface is washed in the cleaning chamber which has water supply equipment and a number of spray nozzles.

6. OPERATIONAL EXPERIENCE

The time schedule for the design and safety approval, construction and operational period is given in Table 1. The design started in April 1978, and the facility was constructed by February 1982. Various
tests on the feasibility of the facility were carried out before initial operation. The tests involved fuel handling, sealing, heat transfer, shielding and so on. The results proved the suitability of this facility.

The transportation of spent fuel and the operation of the facility began in November 1982. About 1,800 fuel elements (15 tons U), half of the capacity, are stored in the facility now.

7. CONCLUSIONS

The storage facility is located at about 3 km from the JRR-3 building and to the North in Tokai Establishment. 4 years were required from the conceptual design to completion of the facility. The dry environment was chosen because of significant advantages: It is considered to be safe, simpler, very reliable, provides negligible contamination, little waste generation and requires a smaller operating staff compared with wet storage.

We have experience of the dry storage of spent fuel elements for 2 years. Consequently it is found that there is no measurable increase of activities through the air circulatory system. The filters provided in the depression system have shown no significant contamination.

We now believe that spent fuel can be stored in the dry storage facility with high assurance that there will be no long-term deterioration.
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</tr>
</thead>
<tbody>
<tr>
<td><strong>Table 1 Schedule of DSF Construction</strong></td>
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<td>Detail Design</td>
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<td>Safety Evaluation</td>
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<tr>
<td>Licensing Review by NSB</td>
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<tr>
<td>Constructions of Building and Equipment</td>
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<td>Inspection by NSB</td>
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<tr>
<td>Feb.</td>
<td>May</td>
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<tr>
<td>Test Operation</td>
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<tr>
<td>May</td>
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<td></td>
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<tr>
<td>Steady Operation</td>
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<td></td>
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<td></td>
</tr>
</tbody>
</table>
| **Note**: Nuclear Safety Bureau in Science and Technology Agency

**Note**: There are 100 drywells in DSF. Presently 51 canisters are stored. Those canisters contain 1798 MNU spent fuel elements (for 50 canisters) and 95 NSRR irradiated test fuel elements (for 1 canister).
Fig. 1 General arrangement of DSF

A: Holes for JRR-3 MNU Fuel

B: Holes for NSRR Irradiated Test Fuel

unit: mm
Fig. 2 JRR-3 metallic natural uranium fuel
Fig. 3 Detail of Drywell
Fig. 4  Detail of Canister

Detail A
- Plug
- Metallic Packing
- Flange
- Spring

Detail B
- Handling Head
- Closure Lid
- Fuel Element
- Inner Case
- Body
- Bottom Plate

Unit: mm
monitor

G : gas monitor

D : dust monitor

P : humidistat

Fig. 5 Monitoring system of DSF
Max. 36 MNU spent fuel elements per inner case.
Total 1798 fuel elements = 50 inner cases (50 canisters)

Fig. 7 Procedure Flow in Hot Laboratory
SESSION 4

DRY STORAGE ECONOMICS, STANDARDS AND LICENSING

Chairman: K.P. Wagstaff
Atomic Energy Control Board
Comparisons are drawn among four possible methods of dry storage of spent nuclear fuel; i.e., storage cask, drywell, concrete silo, and closed-cycle vault. A common set of assumptions is used for all cases, and the results of sensitivity analyses on the principal parameters are presented.

The effects of rate of addition of fuel to the storage facility and of total capacity of the storage system are evaluated with respect to the unit cost of storage, and the incremental cost of extended storage times is examined.

The results provide a basis for comparing the costs of alternative dry methods of storing spent fuel at reactors. Basic parameters selected were:

- Total Storage Capacity: 50, 100, 500, and 1000 MTHM
- Loading Rate: 20 and 40 MTHM/year
- Storage Time: Up to 40 years
1. INTRODUCTION

The spent fuel storage problems being encountered by nuclear utilities have received considerable publicity, both in this Workshop and in reports from various organizations in the field. A number of methods have been suggested for dealing with this problem, the most economical of which is to rerack the spent fuel pool for higher density storage of the spent fuel. This method has been used extensively, but it is ultimately limited in the extent to which it can alleviate the problem. This limitation can be extended by the expedient of consolidating the fuel, that is by disassembling the fuel bundles and close packing the fuel rods in a package of essentially the same cross-sectional area as the original fuel assembly. This expedient is, however, not feasible at those reactors whose spent fuel pools have structural limitations which render them incapable of accepting the higher mass loadings which result.

Current estimates of future spent fuel discharges and reactor storage capacity project an excess of fuel over that which can be stored in existing reactor facilities, assuming maximum reracking of existing pools and continuation of the present trans-shipment of spent fuel between reactors on the same system. Table 1 shows a recent estimate of the expected excess storage capacity required (US DOE, 1984).

This excess can be accommodated by the addition of at-reactor storage facilities. Such facilities may be an independent spent fuel pool (ISFSI), or one of the dry storage modes which have been proposed, including the dry vault facility or one of the modular systems. The latter are represented by metal casks, drywells, and silos (concrete casks). The modular systems are believed by many to have one major advantage over the possible alternatives: the required additional storage capacity can be added as it is needed, permitting the capital costs to be spread over the rest of the life of the reactor plant, rather than being incurred in total at the beginning of the storage requirement. Dry vault storage capability can also be added in
increments, but the practical minimum size of a capacity increment is substantially larger than in the case of the modular systems. The utilization efficiency of the modular units is thus quite high, and they are totally passive systems.

The objective of the study summarized here was to provide a basis for comparing the alternative dry storage methods on the basis of a uniform set of assumptions and using a constant method of calculating costs. It should be emphasized that the capital costs are not based on a complete engineering design, but are based on a fairly detailed evaluation of the capital cost elements involved in each storage system, and estimation, using generally accepted cost factors, engineering judgement, or supplier quotations, of the aggregate capital cost. The operating costs were normalized to parameters of storage capacity, loading rate, and storage time; specific values were assigned to the parameters and were selected to represent a one-unit nuclear power station and a two-unit nuclear power station.

2. BASIC ASSUMPTIONS

The storage units assumed were the metal cask exemplified by the REA 2023 shown in Figure 1, a conventional steel-lined drywell illustrated in Figure 2, a concrete cask or silo, as illustrated by Figure 3. The dry vault selected as the basis of comparison is a modular system employing closed cycle cooling, with each module having a capacity of 45 PWR spent fuel assemblies. The general concept is similar to the ElectroWatt/GA MODREX system, employing heat pipes as the principal cooling means, but differs in that the modules are three-cavity rectangular concrete blocks, and the method of loading employs a transportable cask to transfer from the reactor pool to the storage module.
The estimating procedures used were based on those developed by E.R. Johnson Associates, Inc. (JAI) for estimating and comparing costs of storage in Monitored Retrievable Storage facilities (MRS)\(^{(2)}\) and in the Federal Interim Storage installation (FIS) (US DOE, 1983), and were appropriately modified for the case of at-reactor storage. The costs assume storage of PWR spent fuel, ten years out-of-reactor; mass of the assemblies corresponds to the standard Westinghouse 17 x 17 assembly, with approximately 0.55 kW/assembly heat generation. Intact assemblies were assumed in all cases, but one case was developed to show the impact of spent fuel consolidation. The emplacement rate (the rate at which fuel is transferred into the storage) was assumed to be 20 MTHM* for a single reactor station, and 40 MTHM for a two-reactor station. These assumptions presume an approximately 18-month refuelling cycle with an average annual emplacement of 44 assemblies or 88 assemblies for the two cases.

Fuel is placed directly in casks in the metal cask storage system. For the silo, drywell, and vault systems, however, each fuel assembly is placed in a canister in the reactor spent fuel pool before introduction into the storage module; this is necessary in order to meet the requirement for double containment under 10CFR72. The canister is assumed to be part of the storage module and its cost is included in the module cost; the cost of encapsulation, however, is not included. Spent fuel is transported from the fuel pool to the storage area by an appropriate transporter and shielded container. A commercially available straddle carrier designed for use around marinas can be readily and economically adapted for heavy cask transport (up to 350 tons). The transporter and a transport cask, if required, are included in the storage area cost (as transfer equipment) and the differences in cost of this equipment for each type of storage reflect the special requirements for each storage module. For example, the drywell and the vault storage require a top load/bottom unload shielded transport cask, and the silo may also require the same. Only one transporter is required.

\*MTHM is metric tons heavy metal
All cost estimates are in 1983 dollars. It was assumed for purposes of normalizing the estimates for each case that the design, engineering and licensing costs are incurred in 1984 and construction-related expenses are evenly distributed between 1985 and 1986. For the cases of modular units, it is assumed that the storage modules are purchased in the year prior to their use; thus the annual module and emplacement costs are one year out of synchronization. The decommissioning costs assume that the fuel can be removed from the storage area at a rate determined by the turnaround time for the single transporter on a 24 hour/day operating basis. In all cases except the drywell, the fuel can be removed within a one-year period; the limited capacity of the transporter in the drywell case (one canister with one assembly per trip) results in removal times of two years and four years, respectively, for the storage installations of 500 and 1000 MTHM capacities. It was further assumed that all stored fuel would be returned to the reactor spent fuel pool for transfer to off-site shipping containers; the costs of transferring the fuel into shipping casks is not included, and would not be necessary if the storage casks were rated for both storage and shipping. A typical spending schedule is shown in Table 2, for the 500 MTHM metal cask storage case. Details of the capital cost components for this case are presented in Table 3.

3. DESCRIPTION OF MODULES

3.1 CASKS

The costs and spent fuel loading are based on the Ridihalgh, Eggers and Associates REA-2023 storage-only cask. Based on a quotation from the manufacturer, the delivered cost per cask is estimated to be $765,500; inasmuch as 10-year cooled fuel is assumed, no fins or other cooling accessories are deemed necessary. Cask fuel loading is 24 PWR fuel assemblies. The casks are set on concrete stands or pads, on the REA cask cradle, in a fenced, secure storage area.
3.2 DRYWELLS

The drywell is of conventional design, fabricated from Schedule 40 pipe sections with a total length of approximately 21 feet and a diameter in the lower portion of 16 inches. The liner is grouted in place after being set in a bored hole in the ground. No special ground preparation is assumed, although it is recognized that, in some locations, it may be necessary to construct an engineered berm in order to raise the level of the stored fuel above the ground water table. An areal heat loading of 60 kW/acre is assumed, leading to drywell spacing of 19 feet center-to-center. One canistered PWR assembly is stored in each drywell. The estimated unit cost of the drywell is $18,405.

3.3 SILOS

A sealed, concrete cask approximately ten feet in diameter and 22 feet high, with a 45 inch diameter lined cavity is assumed as the silo storage module. It will hold four canistered PWR assemblies, and is estimated to cost $66,000. Canistered spent fuel is assumed to be transferred in a top loading/bottom unloading transporter cask. If a crane with adequate lift capability is available at the reactor, and if the silo is provided with a suitable exterior waterproof coating, it may be possible to load the silos in the reactor pool in the same fashion as the metal casks. This would result in some reduction in the total costs. As with the metal casks, the silos are stored on concrete pads in a secure storage area.

3.4 VAULTS

The closed cycle vault storage system assumed for purposes of this estimate employs a simple modular storage unit of reinforced concrete, placed at grade, and contains three cavities for storing spent fuel canisters. Each cavity is capable of holding fifteen canisters, for a total loading per module of slightly over 20 MTHM. Assuming nuclear grade concrete for all structures, the cost per module is $1,257,482,
with a cost of $777,485 for the end shielding units. If ordinary structural reinforced concrete is assumed, these costs drop to $986,379 and $353,988 each, respectively. Unit concrete costs for nuclear grade and structural grade are $1885 and $628 per cubic yard in place, respectively. Because the shielding of an individual module is insufficient to reduce the external radiation to an acceptable level at the long side of the module, it is assumed that one end shield and two modules are provided in the initial installation, the costs of which are included with the storage area costs; the remainder of the modules required are installed as needed at a rate that assures that there are always two empty modules ahead of the one in use. The final increment is the end shield.

4. OTHER COST FACTORS

4.1 LICENSING

An independent, at-reactor storage facility must be licensed under 10 CFR, Part 72. Fees for a license under this Part are set at $405,350, based on the latest publication of 10 CFR 170 (10CFR170.31(1)(H)). The cost to the utility for preparation and prosecution of the application is assumed to be $500,000; an additional cost for amending the license at the end of the storage period to permit decontamination and decommissioning of the storage installation is indeterminate (the Part 170 Fee Schedule lists the fee for this activity as "Full Cost"), but the assumed D&D costs should be adequate to cover this cost. The licensing costs are included in the storage area capital costs.
4.2 **DECOMMISSIONING**

Decommissioning costs at the end of the storage period are assumed to be 20 percent of the combined costs of the storage area and the storage modules; canister costs are not included, as the fuel would be shipped off-site in the canisters. To provide a uniform basis for comparison, it was assumed that none of the storage modules would have any salvage value, and that there would be some cost involved in disposing of them. It is possible, of course, that the casks or silos could be used at another storage facility, and if so, that the utility would incur no costs for their disposal.

The costs of decommissioning are treated as an operating expense.

4.3 **OPERATING COSTS**

The operating costs include the labour and utility (primarily electrical) expenses to emplace the spent fuel in the storage modules, maintain the storage facility, and remove the spent fuel at the end of the storage period. Electricity is also required to power the physical security installation, including lighting, alarm systems and communications. Labour costs are based on the following assumptions as to the times required to complete an emplacement cycle:

<table>
<thead>
<tr>
<th>STORAGE MODE</th>
<th>HOURS/EMPLACEMENT</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cask</td>
<td>8</td>
</tr>
<tr>
<td>Silo</td>
<td>16</td>
</tr>
<tr>
<td>Drywell</td>
<td>10</td>
</tr>
<tr>
<td>Vault</td>
<td>16</td>
</tr>
</tbody>
</table>

5. **RESULTS**
5.1 CAPITAL COSTS

The estimated capital costs for the four cases of storage installation total capacity for each storage mode are presented in Table 4. The capital costs comprise the following elements:

- Site Preparation
- Roads
- Fencing
- Security Equipment
- Concrete Pads
- Transfer Equipment
- Monitoring Equipment
- Engineering, @ 15 percent
- Licensing Costs
- License Fee
- Contingency, @ 15 percent

5.2 UNIT STORAGE COSTS

The total unit storage costs are summarized in Tables 5 and 6 for the two emplacement rate cases of 20 MTHM/year and 40 MTHM/year, respectively. Each table shows costs as functions of both total storage installation capacity and total storage time. Total storage time is the elapsed time from the first emplacement to the time of the first removal of fuel for shipment off-site. For the case which assumes location at a single-reactor power station, i.e., the 20 MTHM/year emplacement rate, the unit costs range from a low of $77-80/kgU for the 500 MTHM drywell and silo installations, respectively, to a high of $307-341/kgU for the 50 MTHM vault installation. For the higher emplacement rate cases, the unit costs range from $58/kgU for the silo at 1000 MTHM total capacity to $188-204/kgU for the vault at a 100 MTHM capacity.
It is apparent, as might be expected, that the total capacity of the storage installation has the dominant impact on the unit costs for all storage modes. The differences between the drywell and the silo at both capacity levels are sufficiently small to disregard. It is concluded that there is no significant difference in storage costs between drywells and silos at the higher capacities; at lower capacities, the costs of storage in silos are somewhat higher than drywells.

Figures 4 and 5 are plots of the unit costs as functions of storage installation capacity, and Figure 6 is a plot of unit costs as a function of storage time. Figure 4 is based on the unit costs at the fuel emplacement rate of 20 MTHM/year for each of the storage modes at a total storage capacity of 500 MTHM and a storage time of 25 years; Figure 5 shows the corresponding data for the emplacement rate of 40 MTHM/year. Figure 6 shows that storage time has a relatively small impact on the total unit cost of storage.

5.3 SENSITIVITY CASES

In order to show the effect of variations in the assumptions relating to individual cost factors on the unit storage costs, several examples were calculated using as the base case the 500 MTHM storage installation loaded at 20 MTHM/year. The same cost program was used but the specific element costs were varied. Figure 7 shows the effect of variations over a factor of three in the cask capital cost and the cask operating costs on the resulting unit storage costs. It will be noted that the operating cost variations have a negligible effect on total unit costs, while the cost of the cask itself has an almost linear effect - a not unexpected result. Figure 8 shows the effect of varying the cost of the concrete in the vault structures from the value of $1885/cubic yard for nuclear grade concrete down to a value of $628/cubic yard for high class structural grade concrete; it can be seen that there is a significant effect on the total storage cost.
6. **IMPACT OF FUEL ROD CONSOLIDATION**

In order to demonstrate the impact on the calculated unit storage costs of disassembling the fuel and consolidating the fuel rods in a compact array in canisters of approximately the same size as the original fuel assembly, one case of cask storage in an installation of 500 MTHM capacity was calculated assuming an emplacement rate of 20 MTHM/year of consolidated fuel. The scrap was assumed to be stored in the cask with the fuel rod packages and in the same type canisters. The results are presented in Table 7. Inasmuch as the costs of consolidation are not included in the calculation, the net savings will be less than those shown in Table 7, by an amount equal to the cost of consolidation. These costs, based on previous estimates for a comparable situation (E.R. Johnson Associates, 1981) are likely to be in the range of $19-20/kgU, including the cost of the capital equipment required and consolidation operations during the period fuel is being emplaced. The costs of consolidation are highly influenced by the emplacement rate and the capacity of the storage installation; on the same basis for the 40 MTHM/year rate and 1000 MTHM storage capacity, the costs of consolidation would be in the range of $11.50 - 12.00/kgU.

Separate disposal of the scrap fuel assembly hardware, e.g., by shallow land burial in one of the commercial sites, would increase the apparent savings above those shown in the table, but again, the net savings would be less, by an amount equal to the cost of consolidation plus the cost of disposal, than that shown in Table 7. Other benefits of consolidation, not reflected in the costs of Table 7, are the decrease in shipping costs and the decrease in the number of packages for final disposal. Under the terms of the Nuclear Waste Policy Act of 1982, these advantages would not accrue directly to a U.S. utility, as they represent reductions in the cost to the Department of Energy for the ultimate disposal of the spent fuel.
REFERENCES


TABLE 1

POTENTIAL EXCESS REQUIREMENTS
FOR SPENT FUEL STORAGE

<table>
<thead>
<tr>
<th>Year</th>
<th>Annual PWR (MTU)</th>
<th>Annual PWR (MTU)</th>
<th>Cumulative Total MTU</th>
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<td>-</td>
<td>1</td>
</tr>
<tr>
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<td>27</td>
<td>-</td>
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</tr>
<tr>
<td>1987</td>
<td>67</td>
<td>23</td>
<td>118</td>
</tr>
<tr>
<td>1988</td>
<td>144</td>
<td>-</td>
<td>262</td>
</tr>
<tr>
<td>1989</td>
<td>158</td>
<td>35</td>
<td>455</td>
</tr>
<tr>
<td>1990</td>
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<td>168</td>
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68750
**TABLE 2**

**ESTIMATED SPENDING SCHEDULE FOR STORAGE OF 46 CASKS (500 MTU)**

($000, 1983$)

<table>
<thead>
<tr>
<th>Year</th>
<th>Stor Area Costs</th>
<th>Module Costs</th>
<th>Operat Costs</th>
<th>Mon Costs</th>
<th>Total Costs</th>
<th>Disc. Total Costs</th>
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<tr>
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<td>1,15</td>
<td>4,271</td>
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<tr>
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<td>80</td>
<td>35</td>
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<td>4,271</td>
<td>65</td>
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</table>

Total 0 1,858 36,650 8,682 0 945 48,135 34,953

* - Decommissioning cost

Unit Cost = 94.99 $/KgU  
Discontinued MTU = 367.95  
Storage Period = 25 Years  
Annual Emplacement Rate = 20 MTU/Year
### TABLE 3

**ESTIMATED INITIAL CAPITAL COST FOR CASK STORAGE AREA FACILITY**

($000, 1983)

<table>
<thead>
<tr>
<th>Item</th>
<th>Cost</th>
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</thead>
<tbody>
<tr>
<td>Storage Area Capacity (MTU)</td>
<td>500</td>
</tr>
<tr>
<td>Number of Casks</td>
<td>46</td>
</tr>
<tr>
<td>Cask Unit Cost</td>
<td>797</td>
</tr>
<tr>
<td>Total Cask Cost</td>
<td>36,639</td>
</tr>
<tr>
<td>Site Preparation</td>
<td>21</td>
</tr>
<tr>
<td>Roads</td>
<td>31</td>
</tr>
<tr>
<td>Fence</td>
<td>31</td>
</tr>
<tr>
<td>Security Equipment</td>
<td>74</td>
</tr>
<tr>
<td>Concrete Pads</td>
<td>49</td>
</tr>
<tr>
<td>Transfer Equipment</td>
<td>340</td>
</tr>
<tr>
<td>Monitoring Equipment</td>
<td>72</td>
</tr>
<tr>
<td>Engineering (Facility) (15%)</td>
<td>93</td>
</tr>
<tr>
<td>(Cask)</td>
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</tr>
<tr>
<td>Licensing</td>
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</tr>
<tr>
<td>License Fee</td>
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</tr>
<tr>
<td>Contingency (15%)</td>
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</tr>
<tr>
<td>Total</td>
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<tr>
<td>Storage Type</td>
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<tr>
<td>--------------</td>
<td>---------</td>
</tr>
<tr>
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<tr>
<td>Vault</td>
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</table>

*a*Includes storage area, modules and canisters
### TABLE 5

**DRY STORAGE OF SPENT NUCLEAR FUEL**

**TYPE STORAGE VS STORAGE PERIOD**

**UNIT COST ($/kgHM)**

(Delivery Rate: 20 MTHM/Yr)

<table>
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<th>Capacity (MTHM)</th>
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<th>25</th>
<th>40</th>
</tr>
</thead>
<tbody>
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<td><strong>Cask</strong></td>
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</tr>
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<td>$147.65</td>
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<tr>
<td>100</td>
<td>113.06</td>
<td>116.01</td>
<td>118.21</td>
</tr>
<tr>
<td>500</td>
<td>-</td>
<td>94.99</td>
<td>93.45</td>
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<tr>
<td><strong>Silo</strong></td>
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<td></td>
</tr>
<tr>
<td>50</td>
<td>236.96</td>
<td>256.34</td>
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<td>-</td>
<td>80.98</td>
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<td><strong>Drywell</strong></td>
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<td></td>
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</tr>
<tr>
<td>50</td>
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<td>185.57</td>
<td>195.57</td>
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<td>100</td>
<td>118.92</td>
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<td>129.73</td>
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<td>-</td>
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<td><strong>Vault</strong></td>
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<td>Capacity (MTU)</td>
<td>Years Storage</td>
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</tr>
<tr>
<td>1000</td>
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</tr>
<tr>
<td>1000</td>
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</tr>
</tbody>
</table>

(Delivery Rate: 40 MTHM/Yr)
MONITORING TUBE

WASTE CANISTER CAVITY

SILO DESIGN

(From DOE/RL-34-2)

FIGURE 3

UNIT COST AS A FUNCTION OF STORAGE CAPACITY, 20 MTU ANNUAL LOADING RATE

FIGURE 4
**FIGURE 5**

Unit Cost as a Function of Storage Capacity, 40 MTU Annual Loading Rate

**FIGURE 6**

Unit Cost as a Function of Storage Period at Two Loading Rates

**FIGURE 7**

Effect of Changes in Cost Factors on Unit Storage Costs

**FIGURE 8**

Effect of Vault Concrete Cost on Unit Storage Costs
STATUS OF US STANDARDS FOR SPENT FUEL STORAGE ACTIVITIES*

by John A. Nevshemal
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Iowa 52761, U.S.A.

ABSTRACT

This presentation covers the US industry consensus standards effort in the area and handling of spent commercial LWR fuel. There are four standards involved. All the standards are under auspices of the American National Standards Institute (ANSI) with the American Nuclear Society (ANS) taking the role of manager. One standard covers the siting of an away-from-reactor storage facility, two are concerned with the design of the facility, wet/pool concept and the various dry concepts. The fourth standard concerns the process of fuel assembly rod consolidation. The main point of discussion is the current status of these standards plus the future schedule. Also discussed are various side issues that have lead to the success of this program.

*Presented by J.B. Moegling (T.V.A.) at the Workshop 57030
This presentation deals with the development of US industry consensus standards for the storage of spent nuclear fuel. The main focus will be on those standards that deal with the facilities and equipment required for the safe interim to long term storage of spent commercial LWR fuel. The facilities involved are all away-from-reactor or independent of nuclear facilities. This does not mean that the facility must be located on a separate site but it does require a level of independence which is based in the philosophy of one facility not (operationally) adversely affecting the other. The purpose of this presentation will be to provide you with an understanding of the current status of the development of these standards and some of the important issues involved. Also, discussed will be the goals of the effort, regulatory involvement, the makeup of the various Working Groups and timeliness of the efforts. It should be pointed out that the opinions expressed in this presentation are those of the author and that they have been developed over many years of involvement in the nuclear standards effort. This involvement encompasses the development of seven standards, all of which are concerned with the handling and storage of nuclear fuel. For three of these standards (ANS-57.7, ANS-57.9 and ANS-57.10) this author served as chairman of the Working Group; the other four as a contributing member of the Working Group. All the standards have been and are being developed under the auspices of the American National Standards Institute (ANSI) with the American Nuclear Society (ANS) acting as the manager of the effort. The specific standards that will be discussed are: ANSI/ANS-2.19 Guidelines for Establishing Site-Related Parameters for Site Selection and Design of an Independent Spent Fuel Storage Installation (Water Pool Type); ANSI/ANS-57.7 Design Criteria for an Independent Spent Storage Installation (Water Pool Type); ANS-57.9 Design Criteria for an Independent Spent Fuel Storage Installation (Dry Concept) and ANS-57.10 Design Criteria for the Consolidation of LWR Spent Fuel.
The goal of any standard effort in the nuclear industry is multi-faceted and goes beyond just providing the engineer with the criteria and requirements that he needs to design the facility or equipment. Some of the side issues are the following:

- Promote a level of standardization;
- Identity and/or resolve key technical issues;
- Establish the position of industry prior to regulatory involvement as part of licensing application; and
- Streamline the licensing process through USNRC endorsement of the standard.

There is a school of thought that industry should only write standards after several of the facilities have been designed, built and licensed. This opinion is not shared by this author. In fact, this author feels that this is maybe a waste of time for all concerned because the requirements have already been fought and established in the regulatory arena. This results in the members of the Working Group to have a tendency to represent existing designs. This understandably results in the requirements of the standard being watered down through an attempt to reach consensus. Let me now return to the main issue at hand, which is the status of those standards previously mentioned.

The siting criteria standard ANSI/ANS-2.19 achieved industry concensus and approval by the American National Standards Institute approximately four and a half years ago. The standard has not been endorsed by the US Nuclear Regulatory Commission. The reason for the lack of endorsement is not that it is in conflict with accepted USNRC practice, but that it allows an alternate method for determining the design earthquake. Also it establishes site selection criteria that was based in a radiological risk assessment for the various natural and low
probability manmade phenomena. It is intended that if this criteria is followed in selecting a site, the designer is relieved of designing for the associated events due to extremely low risk.

The regulation that covers away-from-reactor storage facilities or as they are now known, Independent Spent Fuel Storage Installation (ISFSI), is Part 72 of Title 10 of the Code of Federal Regulation. This regulation had not been issued at the time that ANS-2.19 was under development. In fact the development of both ANS-2.19 and Part 72 took place over the same time period. It was a disappointment to the ANS-2.19 Working Group when Part 72 did not allow for the endorsement of the standard, even though the requirements in the regulation are included in the standard. It is my hope that the position in ANSI/ANS-2.19 will be considered for incorporation in any revision of the regulation.

The need for ANSI/ANS-2.19 was identified by the Working Group that was developing the design criteria for the water pool ISFSI. This Working Group, known as ANS-57.7 approached the American Nuclear Society standards governing body known as NUPPSCO (Nuclear Utility Power Plant Standards Committee) with the need for a standard on siting. The effort was assigned to ANS-2 which resulted in the simultaneous development of both standards. In order to establish communication between the two Working Groups (ANS-2.19 and ANS-57.7) the chairman and one other member of ANS-57.7 became members of ANS-2.19. As mentioned previously, the federal regulation covering spent fuel possession and storage facilities (10 CFR Part 72) was also under development at this time. Interface with that effort was also established through USNRC membership on both Working Groups. It is this author's opinion that USNRC membership or involvement in Working Groups is quite valuable and helpful in the development of industry standards.
The status of ANSI/ANS-57.7 will now be discussed. The minimum age of the spent fuel allowed to be stored in a facility covered by this standard was set at one year after discharge from the reactor core. This proviso results in a dramatically reduced hazard level. The reason is twofold: (1) the decay to negligible levels of many fission products; and (2) the reduction in decay heat. The one year age criteria was also incorporated into Part 72. ANSI/ANS-57.7 has both industry approval and USNRC endorsement. The standard was used as the basis for the relicensing of the General Electric facility in Morris, Illinois. This facility is now operating as an Independent Spent Fuel Storage Installation. The ANS-57.7 effort is a fine example of having the industry position established prior to regulatory involvement through the licensing process. It also demonstrates that it is possible for industry to have an input to regulatory requirements at a time when their roles are non-adversary (i.e. prior to a licensing action). This ultimately results in streamlining the licensing process because both the applicant and the regulatory body are working to known and mutually accepted requirements.

The next standard that will be discussed is ANSI-57.9. This standard presents the design criteria for the various dry storage concepts. There are three basic storage concepts that are considered as dry. They are: (1) vault or canyon; (2) silo or cask; and (3) drywell or caisson. Since this standard applies to the handling and dry storage of spent fuel in an independent facility, it presents criteria for all the various support areas that are needed, along with those required for the three storage concepts. The current status of this standard is one of having been balloted on by the consensus body, NUPPSCO. At the time of drafting this presentation, all the negative ballots have been upgraded to approved. It is expected that the standard will be officially approved by ANSI and published. Dry storage of spent commercial LWR fuel is new to the United States. Several other countries have developed and are operating dry storage facilities, specifically Canada and the United Kingdom uses dry storage concepts. The ANS-57.9
Working Group was able to incorporate this experience into the standard. This was achieved by having representation from these two countries on the Working Group.

The ANS-57.9 Working Group is outstanding in many ways. Besides having the best expertise available internationally, it had balanced representation from within the industry. It also had review lines of communication with other standard committees. During the development of ANS-57.9, several areas of needed research were identified by the Working Group. These included the behaviour of \( \text{UO}_2 \) in limited and unlimited air environments. The need for this research came about when the Working Group searched for data to support the temperature limit at which the fuel can be inserted into a storage concept that uses air. The concern is the transformation of \( \text{UO}_2 \) to \( \text{U}_3\text{O}_8 \) through oxidation. The \( \text{U}_3\text{O}_8 \) phase is accompanied by a rapid volumetric increase. This volumetric increase could result in the failure of the cladding of the irradiated fuel. The transformation to \( \text{U}_3\text{O}_8 \) is theorized to be a time-at-temperature reaction with cumulative damage. There is sufficient unirradiated fuel experimental data to support this theory. There is insufficient data on irradiated urania to establish a temperature limit that will assure \( \text{U}_3\text{O}_8 \) is not formed during storage lifetime. What little data exists does indicate that dry storage concepts using air are, indeed, feasible. The need for this data and design methodologies using it has been the subject of several past meetings hosted by the US Department of Energy and the US Nuclear Regulatory Commission. The subject will also be discussed at this workshop. It is hoped that the research will produce useable engineering data in the near future. Research is also needed in the area of reactor-induced cladding defects as to how their size and shape affects the diffusion of the storage environment into the fuel.
The last standard in the area of spent nuclear fuel handling and storage is the one concerned with rod consolidation which is ANS-57.10. The Working Group of this standard has begun the development effort. In fact a complete review of the first draft took place earlier this week here in Toronto. The Working Group which also has international involvement from West Germany, met for the first time approximately eighteen months ago. A subsequent meeting was held in November, 1983 at which a first draft of the standard was produced. A couple of the main issues to be addressed in this standard are the concern for criticality and the potential for stuck or broken fuel rods. Criticality control for consolidated rods is based on maintaining an undermoderated condition whereas with whole fuel assemblies, the condition to be maintained is overmoderation. Demonstrations of rod consolidation indicate that stuck rods may be a routine event. The standard intends to deal with rod consolidation as a process that could take place in either a nuclear facility or a storage installation and could be performed either wet or dry. The necessary interface criteria for the process is also being developed as part of the standard. The schedule for the development of the standard is to have a final draft for ballot by mid-1985. The approval of ANS-57.10 will complete the set of similar standards that are concerned with the storage of spent commercial LWR fuel.

The entire effort has been greatly helped by financial support that has been afforded each of the Working Groups. The support provided for meeting and secretarial expenses. It also helped attract the best available talent to the Working Groups and aided in keeping the development effort on a 2 to 2-1/2 year schedule. This is about half of what standard development schedules have been. Industry, the American Nuclear Society and the USNRC should be complimented for this truly outstanding effort in meeting the standard needs of this most crucial nuclear issue.
Design guidelines are being prepared for a dry store for spent AGR fuel. To ensure a consistent approach to safety, the guidelines derive from criteria that have been prepared by the Health and Safety Department of the C.E.G.B. for nuclear power reactors.

A vault concept has been selected for detailed engineering and safety studies.
1. INTRODUCTION

The C.E.G.B. has been considering the option of dry storing AGR fuel for about 5 years and has chosen and developed a concept for which a pre-construction safety report (PCSR) is presently being prepared. It should be pointed out that the C.E.G.B. already has an agreement with British Nuclear Fuels to reprocess about 25% of the fuel arising from the current AGR programme. As yet there has been no decision to change from reprocessing to dry storage for the remaining fuel but dry storage is recognized as an alternative to early reprocessing.

This presentation describes briefly the areas of responsibility of the authorizing bodies in the UK, explains how the design safety requirements for the dry store are being developed and gives examples, and lastly outlines the features of the storage vault which it is believed will allow the safety objectives to be met.

Nuclear site licenses in the UK are issued by the Health and Safety Executive (HSE) in accordance with the Nuclear Installations Acts of 1965 and 1969. Such a license is only granted after the design of the installation has been subjected to detailed scrutiny by H.M. Nuclear Installations Inspectorate (NII), an arm of the HSE. Authorizations for off-site discharges of radioactive substances are issued in accordance with the Radioactive Substances Act of 1960 by the Department of the Environment (DOE) and the Ministry of Agriculture Fisheries and Food (MAFF) in England and by the appropriate Secretary of State in Wales and Scotland.
The authorizing departments also require assurances that the radioactive waste management practices at the installation have been optimized and that discharges are as low as reasonably practicable and are compatible with the national waste management policy. The authorizing departments are also involved in assessing the environmental impact of the installation. In addition the UK must obtain the approval of the European partners under Article 37 of the Euratom treaty.

2. DESIGN CRITERIA

The NII does not issue detailed regulations and standards which have to be met before a license for construction can be issued. Instead the responsibility is placed on the license applicant to develop a safety case from basic principles. This safety case is presented in the PCSR and it subjected to a rigorous examination principally by the NII. To assist this process the NII has issued its Assessment Principles for Nuclear Power Reactors and Assessment Principles for Nuclear Chemical Plant.

Within the C.E.G.B. it is the independent Health and Safety Department (HSD) which is responsible for formulating and providing advice on safety policy and for acting as the formal channel of communication with the NII and authorizing departments.

Over the last 10-15 years increasing attention has been given worldwide to nuclear safety and a trend towards escalating safety requirements has been apparent. To assist designers against this background the HSD prepared a set of Design Safety Criteria for C.E.G.B. Nuclear Power Stations which consolidated its own experience and worldwide developments in safety. The fundamental principles of this document are shown in Table 1.
In order to achieve the objective of a balanced design in terms of safety, the HSD encourages in this document the use of probabilistic risk assessments although it is recognized that it may not always be possible to provide the necessary degree of assurance in the numerical predictions for all items of plant. In such cases, the HSD is prepared to accept non-numerical safety claims based on engineering arguments. Safety assessments by HSD, however, continue to be based upon the deterministic approach using judgement and experience and an appreciation of all the engineering features designed into the plant.

A detailed and designer oriented guide to the implementation of the criteria for reactor plant, the Design Safety Guidelines, has been prepared by the Generation Development and Construction Division (GDCD) of the C.E.G.B.

Neither the Design Safety Criteria nor the Design Safety Guidelines are mandatory but they are accepted by the NII on the basis that their application should lead to an acceptable plant design.

Whilst the Design Safety Criteria relate specifically to nuclear power stations it was recognized that a dry fuel storage facility should as far as possible be consistent with them and that the recommended approach to safety should be used for the dry store. Accordingly a set of Design Safety Guidelines were presented in the Dry Store Preliminary Safety Report (PSR) and these are being developed for presentation in the PCSR.

The reactor Design Safety Criteria and Guidelines set out the radiation dose target for operating staff and the public for normal operating conditions and probability criteria for accidental releases to the atmosphere. These have been adopted with only minor changes for the Dry Store, Tables 2, 3, 4 and 5.
Also set out in the reactor documents are the types of internal and external hazard which must be considered in the safety assessment. These will be included in the Dry Store Guidelines where they are relevant. Internal hazards will include fires and dropped loads and external hazards will include consideration of earth tremors, extreme winds, site flooding and aircraft impact for example. As yet no site has been nominated for the dry store but it is expected that no specific provision will be necessary for the last two items.

Finally, the reactor documents detail the engineering criteria and it is here that they are least relevant to the dry fuel store since that is an unpressurized system with low heat generation fuel. Nevertheless criteria covering such aspects as reliability requirements for essential systems, segregation of plant, inspection, contamination control, waste management, quality assurance etc. can sensibly be applied to a dry store and they will be included in the Dry Store Design Safety Guidelines.

3. AGR DRY STORE CONCEPT

The requirement on the AGR dry store is that it should be able to accept 100 day cooled fuel at the rate generated by the seven AGR power stations in operation or under construction and store it for a period between 50 and 100 years. The total amount of fuel stored should be about 7000 teU.

After careful consideration of the design options a modular vault system with natural draught cooling was selected. The design is jointly owned by the C.E.C.B. and the National Nuclear Corporation and has been the subject of considerable engineering and development effort. The main design features are presented here.
An AGR fuel element consists of a bundle of 36 x 15 mm dia. stainless steel clad fuel pins in a graphite sleeve whose overall dimensions are approximately 1000 mm x 240 mm dia. Elements discharged from a reactor are stored in a pond for at least 100 days before dispatch from the station. The decay heat generation in a fuel element varies with mean fuel irradiation and axial position in the core but for Dry Store design purposes mean and maximum element heat generation rates of 500 W and 750 W are taken. Fuel elements are taken to the Dry Store in normal transport flasks and are then removed and dried and placed four high in carbon steel containers. The containerization process is an essential component in the safety case for long term storage and considerable effort is made to check that the fuel has been correctly dried and the container lid correctly welded. The lid weld is helium leak tested and the container is twice evacuated and backfilled with argon.

The final container closure consists of mechanical seal backed by a closure weld. Drop tests with containers have confirmed that they are more than able to withstand the maximum impact.

The storage vault (Figure 1) consists of a concrete matrix of 625 channels into which containers are loaded two high. Air is drawn over the containers by natural convection.

Corrosion control is by re-circulating about 50% of the hot air flowing from the channels to the inlet plenum so that the relative humidity at the channel inlets is always below about 30%. Re-circulation is achieved by entrainment of the warm exit air at the inlet nozzles.

Each vault can hold one years arisings from the current AGR programme and when full has a heat loading of about 1 MW. The fuel pin temperature within the containers is about 270°C on placement falling to about 200°C after 1 year and 150°C after 2 years.
The air discharged from each vault stack is continuously sampled and checked for radioactivity. If a release is detected then each channel is sampled by means of a vehicle operating in the lower plenum and the offending container located and removed. During the period the vault can be switched to forced cooling with the effluent air passing through absolute filters.

Vault internals are subjected to periodic inspection and selected containers are periodically removed for external and internal inspection.

The features of the storage vaults upon which the safety case is based are summarized in Table 6.

REFERENCES

TABLE 1

DESIGN SAFETY CRITERIA FOR C.E.G.B. NUCLEAR POWER STATIONS

Fundamental Principles:

As a result of normal operation of a nuclear power station, no person shall receive doses of radiation in excess of the appropriate limits;

The exposure of individuals to radiation shall be kept as low as is reasonably practicable;

The collective dose equivalent to operators and the general public as a result of the operation of a nuclear installation shall be kept as low as is reasonably practicable;

All reasonably practicable steps shall be taken to prevent accidents;

All reasonably practicable steps shall be taken to minimize the radiological consequences of any accident.
TABLE 2

DOSES IN NORMAL OPERATION

Occupational Doses

(i) The maximum individual effective dose equivalent to any member of the occupationally exposed staff shall not exceed:

1 rem (10 mSv) per annum

(ii) The maximum dose equivalent to the lens of the eye on any individual occupationally exposed person shall not exceed 15 rem (0.15 Sv) per annum and to the extremities shall not exceed 50 rem (0.5 Sv) per annum.
Estimated doses to the public should be derived for:-

(i) liquid effluent discharges
(ii) gaseous effluent discharges
(iii) direct radiation from the plant

and any other existing sources on the site and assessed against a level of 1/30 of the limits given in ICRP Publication 26 for the General Public.
### TABLE 4

ACCIDENTAL RELEASES TO THE ATMOSPHERE

<table>
<thead>
<tr>
<th>Dose Band</th>
<th>Releases giving rise to doses</th>
<th>Total permissible frequency of release p.a.</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>ERL 1000 to ERL 100</td>
<td>$10^{-2}$</td>
</tr>
<tr>
<td>2</td>
<td>ERL 100 to ERL 10</td>
<td>$10^{-3}$</td>
</tr>
<tr>
<td>3</td>
<td>ERL 10 to ERL 1</td>
<td>$10^{-4}$</td>
</tr>
</tbody>
</table>

### TABLE 5

ERLs TO BE USED FOR ASSESSMENTS

<table>
<thead>
<tr>
<th>Organ</th>
<th>Dose rem (Sv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Whole body</td>
<td>10 (0.1)</td>
</tr>
<tr>
<td>Thyroid, lung or other single organs</td>
<td>30 (0.3)</td>
</tr>
<tr>
<td>Skin</td>
<td>100 (1.0)</td>
</tr>
</tbody>
</table>

An ERL is the lower limit for evacuation as specified by the National Radiological Protection Board.
<table>
<thead>
<tr>
<th></th>
<th>Feature</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Containerization of dried fuel in an argon atmosphere.</td>
</tr>
<tr>
<td>2</td>
<td>Leak testing of welded containers prior to placement in a vault.</td>
</tr>
<tr>
<td>3</td>
<td>Passive vault cooling system.</td>
</tr>
<tr>
<td>4</td>
<td>Conditioning of inlet air to avoid storage container corrosion.</td>
</tr>
<tr>
<td>5</td>
<td>Continuous monitoring of effluent air.</td>
</tr>
<tr>
<td>6</td>
<td>Standby vault filter system.</td>
</tr>
<tr>
<td>7</td>
<td>System for location of faulty containers.</td>
</tr>
<tr>
<td>8</td>
<td>In-service inspection of vault structures and containers.</td>
</tr>
</tbody>
</table>
ABSTRACT

Carolina Power & Light (CP&L), the U.S. Department of Energy (DOE), the Electric Power Research Institute (EPRI), and NUTECH Engineers are co-operating in a demonstration of the horizontal dry storage of canistered spent fuel in modular concrete silos at CP&L's H.B. Robinson Plant. This paper outlines the program objectives and major tasks, and describes the Nutech Horizontal Modular Storage (NUHOMS) generic design. In late 1984, the generic design will be modified for the Robinson site demonstration which will utilize the General Electric IF300 shipping cask owned by CP&L as the transfer cask.
1. INTRODUCTION

When commercial nuclear power plants were first built in the United States and elsewhere, it was anticipated that once the nuclear fuel had been used to produce electricity that the spent fuel would be sent to a reprocessing plant to recover reusable portions of the nuclear fuel. Since that time over 80 nuclear power plants have been placed into service in the United States. There are, however, no currently operating reprocessing plants in the U.S. As a consequence, spent fuel is accumulating at the nuclear power plants.

The Nuclear Waste Policy Act of 1982 established 1998 as an operational date for a spent fuel/high level radioactive waste repository. By that time, the Department of Energy (DOE) estimates that the backlog of spent fuel will exceed 40,000 metric tons.

To assist utilities in providing for spent fuel storage until the repository is operational, the Nuclear Waste Policy Act requires DOE to "... establish a demonstration in co-operation with the private sector for the dry storage of spent nuclear fuel at civilian nuclear power reactor sites, with the objective of establishing one or more technologies that the Commission (NRC) may by rule approve for use at the sites of civilian nuclear power reactors without to the maximum extent practicable, the need for additional site specific approvals by the Commission." Accordingly, the DOE issued its Solicitation for Co-operative Agreement Proposal for a License At-Reactor Dry Storage Demonstration Program.

Carolina Power & Light (CP&L), in response to the DOE solicitation, submitted a proposal to demonstrate the NUTECH Horizontal Modular Canage (NUHOMS) system at CP&L's H.B. Robinson site located near Hartsville, South Carolina. The NUHOMS system was developed by NUTECH Engineers for the horizontal dry storage of canistered spent nuclear fuel in modular concrete silos.
This paper outlines the program objectives and major tasks, and describes the NUHOMS generic design. In late 1984, the generic design will be modified for CP&L's Robinson demonstration program.

Successful completion of the demonstration program will result in an economical dry storage alternative beneficial to the nuclear industry in meeting interim spent fuel storage needs until a repository is operational and the government can receive the spent fuel.

2. STORAGE STRATEGY

CP&L, like other utilities, has expanded its Robinson and Brunswick spent fuel storage capacities to the maximum practical limits. Limitations on floor loadings and other considerations prevent any further capacity expansions in the pools of these plants.

While spent fuel storage in metal casks is being developed in other demonstration programs, storage of canistered spent fuel in horizontal concrete silos for ease of placement and retrieval shows promise of being an attractive and potentially lower-cost option. To prove the approach and qualification of a design for use at U.S. reactor sites, a program is required to design, construct, perform thermal and shielding analysis, develop supporting thermal and shielding test data, and verify concrete and cladding temperatures.

3. PROGRAM OBJECTIVES

The objectives of the demonstration program are:

1. Demonstrate the dry storage of spent fuel using large capacity canisters housed in concrete silos.
2. Achieve economic storage through the use of a modular design requiring low capital investment which is favourably competitive with metal casks.

3. Prove licensability of the concept under NRC regulations and pave the way for future faster licensing, as provided in the Nuclear Waste Policy Act, by other utilities utilizing concrete silos.

4. Demonstrate a systems approach to interim storage by providing storage that is independent of future changes in shipping and transportation regulations and requirements and which has possible future repository interface benefits.

5. Develop test data for dry spent fuel storage that are generically applicable and therefore of general use to the entire nuclear utility industry.

4. MAJOR PROGRAM TASKS

The major program tasks can be summarized as follows:

1. Develop a detailed design for the canister, the concrete modules, and associated cask transfer hardware.

2. Perform preparations tests to verify air flow and concrete temperatures.

3. Submit licensing generic topical report and site specific application to obtain Nuclear Regulatory Commission (NRC) approval.
4. Conduct licensed demonstration of storage for a total of approximately nine tons of fuel in three modules (three tons per module).

5. **PROGRAM PARTICIPANTS**

CP&L is the participant who will implement the demonstration program at its Robinson facility.

EPRI, through its External Fuel Cycle Subcommittee of the Systems and Material Task Force, is providing both financial and programmatic support. EPRI's involvement provides representation and participation by the nuclear utility industry, helps to assure generic applicability, helps provide for integration with other dry storage research and development projects being supported by the nuclear industry, and helps provide for the dissemination for the results of the program.

DOE, through its Civilian Spent Fuel Management Program Office managed by the DOE Richland Operations Office, is also providing financial and programmatic support. This is authorized by Section 218 of the Nuclear Waste Policy Act which requires the establishment of demonstration programs for the dry storage of spent nuclear fuel at up to three civilian nuclear power plant sites. DOE's role in the demonstration program includes assistance in areas such as licensing support, heat transfer analysis, structural analysis, data analysis, and technical reviews.

NUTECH Engineers, CP&L's subcontractor who developed the NUHOMS system, will provide engineering, licensing, and other services to implement the demonstration program. NUTECH is also providing financial support to the program in the form of a generic design and licensing services.
6. DESCRIPTION OF THE NUHOMS SYSTEM

This portion of the paper describes the NUTECH Horizontal Modular Storage (NUHOMS) system. The NUHOMS system provides for the horizontal, dry storage of spent fuel in a dry, shielded canister (DSC) which is placed in a concrete horizontal storage module (HSM). The NUHOMS system is designed to be installed at any reactor site or any new site where an independent, spent fuel storage installation is required. Figure 1 shows the primary components of the system.

The spent fuel assemblies are loaded into the DSC which is inside a transfer cask in the fuel pool at the reactor site. The transfer cask containing the filled DSC is removed from the pool and dried. The DSC internals are then dried and backfilled with helium (He). The cask is then placed on a transfer trailer and towed to the storage location. At the storage location the DSC is moved from the transfer cask into the HSM by a horizontal hydraulic ram.

The principal design features of a NUHOMS installation are:

1. Horizontal Movement of the DSC Into and Out-of the HSM

This removes the need for a critical lift of the spent fuel at the storage location (i.e., away from the fuel pool crane) reduces shielding requirements and results in a low profile, impact resistant structure.

2. Shielded Ends on the DSC

This enables the direct handling and monitoring of the canisters at the end positions when the DSC is inside the cask or HSM.
3. Unitized Construction of HSM

This facilitates licensing and commercial production of the HSM and permits spreading of the investment for fuel storage over the time span when storage is required.

4. Natural Circulation Air Cooling

This keeps the maximum fuel rod cladding surface temperature below 350°C.

5. Accepts fuel up to 1 kw/assembly with burnup up to 35,000 Mwe/Mt. This provides for acceptance of most reactor power fuel cooled for at least five years.

6. Helium Storage Atmosphere

This improves heat transfer and prevents cladding and uranium dioxide (UO₂) oxidation.

6.1 STRUCTURAL FEATURES

The HSM is a low profile reinforced concrete structure designed to withstand all normal operating loads, the abnormal loads created by seismic activity, tornados and other natural events and the accidental loads which might occur during operation.

The structural features of the DSC are defined, to a large extent, by the cask drop accident. The canister body and the double containment welds on each end are designed to withstand the drop without breaching containment. Furthermore the DSC internals are designed to withstand a cask drop without any permanent deformation.
6.2 DECAY HEAT DISSIPATION

The decay heat of the spent fuel is removed from the DSC by natural draft convection. Air enters the lower part of the HSM, rises around the DSC and exits through the top shielding slab. The flow cross-sectional area is designed to provide adequate air flow from the draft height of the HSM and the inlet and outlet air temperature differences for the hottest day conditions (i.e., 54.5°C (130°F) inlet and 82.2°C (180°F) outlet).

6.3 SAFETY FEATURES

The principal safety feature of the NUHOMS system is the containment provided by the DSC and the concrete shielding of the HSM. This shielding reduces the gamma and neutron flux emanating from the spent fuel inside a DSC so that the average outside surface dose rate on the HSM is less than 20 mrem/hr. Additional safety features include:

<table>
<thead>
<tr>
<th>Feature</th>
<th>Purpose</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Filling DSC and cask with demineralized water prior to lowering in fuel pool</td>
<td>Prevents contamination of DSC exterior by pool water</td>
</tr>
<tr>
<td>2. Internal shield blocks inside HSM</td>
<td>Reduces scatter dose out of air inlet</td>
</tr>
<tr>
<td>3. External shield blocks on HSM</td>
<td>Reduces scatter dose out of air outlet</td>
</tr>
<tr>
<td>4. Shielding ring between DSC and cask internal wall</td>
<td>Reduces scatter dose from between DSC and cask during canister drying and sealing</td>
</tr>
</tbody>
</table>
5. Shield plugs on DSC Reduces dose during DSC drying, the filling and seal welding

6. Double containment closure Prevents fission gas escape if welds on each end of the DSC rods are breached

These safety features allow the NUHOMS system to be deployed with a minimum of additional, site specific radiation protection and monitoring measures.

6.4 GENERAL SYSTEMS DESCRIPTIONS

The major systems, subsystems and components of the NUHOMS system are shown on Table 1. The following subsections briefly describe the principal systems and components and their operation.

6.4.1 Canister Design

Figure 2 shows the DSC. The DSC is sized to hold seven irradiated PWR fuel assemblies. The main component of construction is the cylindrical stainless steel pipe body.

The internal basket of the DSC is also shown in Figure 2. The basket is composed of seven square cells. Each cell wall contains stainless steel clad boral. This provides sufficient neutron poison to control criticality in the canister during wet loading operations.
Table 1

MAJOR SYSTEMS, SUBSYSTEMS AND COMPONENTS OF THE NUHOMS SYSTEM

Dry Shielded Canister
  Canister Basket
    Square Cells
    Spacer Disk
    Support Rods
  Canister Body
  Shielded End Plugs (1 on each end)
  Steel Sealing Plates (2 on each end)

Horizontal Storage Module
  Dry Shielded Canister Holding Assembly
  Alignment System
  Door
  Air Flow Penetrations

Trailer
  Cask Skid
  Skid Positioning (Vertical and Horizontal)

Transfer Cask
  Cask Body
  Cask Lids (2)
  Cask Drains
  Cask – Horizontal Storage Module Coupling Collar

Structural support is provided by the circular, stainless-steel spacer disks. The basket is designed so that there is one disk under each spacer in the fuel assembly. Longitudinal support is provided by the four support rods which run the length of the canister from one end shield to the other.
The DSC is equipped with two shielded end-plugs so that when the canister is inside the transfer cask or the HSM, the radiation dose is limited and the ends are accessible for handling. The end shields are constructed of two steel plates and a lead shield.

The DSC has double, redundant seal welds at the top and bottom. Two separate stainless steel plates are welded to the DSC body on each end. The bottom plates are welded during fabrication and the top plates after fuel loading, drying and backfilling with He. Also, all connections (drain and air purge lines) will be redundantly sealed. This assures that no single failure will breach DSC. Furthermore, there are no feasible accidents which would breach the main body.

Criticality safety during wet loading operations is maintained through the geometric separation of the fuel assemblies by the basket cells, and by the neutron absorbing material used within the central basket cell walls. Furthermore, criticality safety will be assured by the fission products in the fuel (however, this credit is not considered during analysis).

6.4.2 Horizontal Storage Module

Figure 3 shows the HSM. The HSM is a reinforced concrete structure located on a load bearing foundation. The HSM provides a unitized modular storage location for irradiated fuel. The HSM is constructed from components fabricated from reinforced concrete structural steel and stainless steel. The modules can be poured in place at the storage location or shop fabricated and shipped to the storage location.

The thick concrete top and front of the HSM provide adequate neutron and gamma shielding. Nominal surface doses are less than 20 mrem/hr for the front locations where workers must perform loading and unloading operations.
The thick shield walls are added (i.e., poured at the same time or connected by bolts) to the end modules of a unit of eight to provide shielding on the sides. Sufficient shielding between modules (to prevent scatter in adjacent models during loading and retrieval) is provided by the normal module walls.

The HSM provides fuel cooling by a combination of radiation, conduction and convection. Heat is conducted out of the DSC into the natural convection air flow in the internal passage of the HSM. Heat is also radiated from the DSC to the HSM walls where again the natural convection air flow removes the heat. The air enters at the bottom of the HSM and passes around the DSC and exits through the flow channels of the top shield plate. The passive cooling system of the HSM was designed to assure that peak cladding temperatures are less than 350°C.

The HSMs are independent, passive systems for the dry storage of spent nuclear fuel assemblies. Therefore, the HSMs are safety related and, hence, are designed to ensure that normal operation and credible site hazards do not impair their safety function. To this end, the HSMs are designed to the following loads:

1. Wind and Tornado Reg. Guide 1.76
2. Seismic Reg. Guide 1.60, 1.61, 10CFR72.66
3. Flood 10CFR72.72
4. Snow and Ice
5. Combined Load (dead weight, ACI 349-80 live loads, temperature)

As stated above, the HSMs are placed in service on a load bearing foundation. This foundation, which includes a trench and catch sump, is placed within a fenced, controlled access area. Necessary civil work is required to prepare the storage site for a level foundation and access area.
6.4.3 Transfer Cask

The transfer cask used in the NUHOMS system provides shielding during the DSC drying operation and during the transfer to the HSM. The internal cavity of the cask must be approximately the same size as the external dimensions of the canister (a free sliding fit). Any suitable cask design which meets these requirements and the licensing requirements of 10CFR72 can be used in the NUHOMS system. Indeed, such a cask is currently licensed under 10CFR71 as a transportation cask. This cask, the General Electric IF-300, is fully acceptable for the NUHOMS system. Future, simpler casks specifically designed as a NUHOMS transfer cask could also be used at a potentially significant cost savings.

6.4.4 Transporter

The transporter consists of a low-boy trailer with a capacity of approximately 100 tons. The trailer carries the cask skid and the loaded transfer cask. The trailer is designed to ride as low to the ground as possible to minimize the HSM height. The trailer is also equipped with four hydraulic jacks to provide vertical movement for alignment of the cask and HSM. The trailer is pulled by a conventional tractor truck.

6.4.5 Skid

The skid is similar in design and operation to the present IF-300 skid. The major differences are:

1. There is no equipment in front of the top of the cask when in the horizontal position.

2. The skid is mounted on a roller surface with hydraulic positioner to provide alignment with the HSM. Locking bolts are provided to prevent movement during trailer towing.

3. The entire skid is mounted on a low-boy trailer.
6.4.6 **Horizontal Hydraulic Ram**

The horizontal hydraulic ram is a telescopic, hydraulic boom with a capacity of 10 tons and a reach of 25 feet. The ram can be truck mounted, in which case it is very similar in design and function to the booms on truck-mounted hydraulic cranes presently used at power plant sites.

6.4.7 **System Operation**

The primary operations (in sequence of occurrence) for the NUHOMS system are:

- Cask Preparation
- Canister Preparation
- Cask-Canister Loading
- Canister-Assembly Loadings
- Cask Head Placement
- Cask Lifting
- Cask-Canister Drying
- Canister Sealing
- Helium Filling
- Transporter-Skid Loading
- Transfer
- Cask-Module Preparation
- Module Loading
- Storage
- Retrieval

6.4.7.1 **Cask Preparation**

Cask preparation includes exterior washdown and interior decontamination. These operations are done on the decontamination pad outside the fuel pool area. The operations are standard cask operations and have been performed by numerous utility personnel and others in the nuclear industry. Detail procedures already exist for these operations.
6.4.7.2 Canister Preparation

To the extent practical, the internals and externals of the canister are thoroughly washed. This ensures that the newly fabricated canister will meet existing station-specific criteria for placement in the irradiated fuel pool.

6.4.7.3 Cask-Canister Loading

The empty canister is inserted into the cask. Proper alignment is assured by an internal key inside the cask and the tight (sliding) fit on the DSC.

6.4.7.4 Cask Lifting

The cask and DSC inside the cask are filled with demineralized water. This prevents an inrush of pool water when they are placed in the pool. This will also reduce (if not prevent) contamination of the DSC outer surface by the pool water. The water-filled cask with the DSC inside is then lifted into the fuel pool.

6.4.7.5 Canister-Assembly Loading

Seven assemblies are placed into the canister basket. This operation is identical to the present cask basket loading operation.

6.4.7.6 Cask Head Placement

Cask head placement operation consists of placing the DSC upper end-shield head inside the DSC using the overhead crane. When the DSC shield head is in place the shielding ring is placed around the upper end of the canister (between the DSC outer surface and the cask inner surface). After the canister shielding ring is in place, the cask head is placed on the cask, the cask is raised to the surface, and the cask head bolts secured. The cask head firmly holds the DSC in place while in the cask.
6.4.7.8 Cask-Canister Drying

The cask is first drained, thus removing all the potentially contaminated water which is between the canister exterior and the cask interior. After draining, clean demineralized water is flushed through the canister-cask gap to remove any contamination left on the canister exterior. The gap is then drained and the cask head removed, allowing access to the DSC. The DSC upper shield head is bolted in place forming an airtight seal. Because the DSC has end shields, the upper end of the cask and DSC will be accessible to perform this operation by hand. Next, air lines are connected to the DSC and the water inside the canister is forced out by air pressure. The water, which is removed from the cask and the DSC, is routed to the plant radioactive waste processing equipment. In addition to forcing air into the DSC, the air line is used to draw a vacuum to facilitate drying until the water content meets the design criteria.

6.4.7.9 Canister Sealing

Once the moisture content of the DSC is below the design limit, the upper steel lid plate is seal welded. A circular welding shield is placed between the DSC and the cask internals to prevent damage to the cask. The secondary steel lid plate is positioned and seal welded after the helium filling operation (next subsection). This provides a redundant seal at the upper end of the DSC. The lower end also has redundant seal welds, which were provided and tested during fabrication. This operation provides the double seal integrity of the DSC.

6.4.7.10 Helium Filling

In order to ensure that no fuel and/or cladding oxidation occurs during storage, the canister is filled with helium (He). To accomplish this the air line (which was used for drying) is removed and a portable He gas bottle is connected.
The canister is then filled with the He gas to a pressure of 14.7 psia. After the canister is filled with the inert gas, the filling line is removed and the DSC line is plugged and welded closed. Because the plug at the surface of the upper steel lid is welded in place, the integrity of the air (helium) line and the water line is assured.

6.4.7.11 Transporter-Skid Loading

After helium filling, the cask transport lid is positioned and bolted in place to close the cask for transport to the HSM. The cask is then lifted onto the skid mounted on the transfer trailer. This operation is identical, in principle, to the present cask-skid loading, the exceptions being a new, slightly modified skid and transfer trailer.

6.4.7.12 Transportation

Once loaded and secured, the transfer trailer is towed to the HSM. This movement is completely within the power plant site.

6.4.7.13 Cask-Module Preparation

At the HSM storage area, the cask transport lid and the lower cask cover are removed and the coupling collars are mounted on the cask and HSM. The coupling collars are lead O-rings which are bolted in place. Next, the transfer trailer is backed to within a few feet of the HSM, and the canister shielding ring is removed. The trailer is then backed to within one foot of the HSM and the hydraulic ram which has been extended through the lower cask cover is attached to the DSC.

The transport trailer is then backed into position, so that the coupling collars are in compression. A laser alignment system and the hydraulic skid positioners are used for the final alignment of the canister and module.
6.4.7.14 Module Loading

After the final alignment is completed, the shield door is raised in its slot and pinned in the open position. The canister is then pushed into the HSM by the 10-ton hydraulic ram (located at the rear of the cask).

6.4.7.15 Storage

After the DSC is inside the HSM, the hydraulic ram is released from the DSC and withdrawn through the cask. The transfer trailer is pulled away and the HSM shield door is closed.

6.4.7.16 Retrieval

The hydraulic ram is used to move the DSC into the cask. For retrieval, the cask is positioned as previously described and the DSC is horizontally transferred. All coupling, attachment, alignment, and closure operations are done in the same manner as previously described, but in reverse order. Once back in the cask, the DSC and its cargo of canistered spent fuel are ready for shipment to a repository or other storage location.

7. SUMMARY

The Robinson demonstration program will demonstrate NUTECH Engineer's dry storage technique using horizontal concrete silos. The program will provide data generically applicable to dry spent fuel storage, demonstrate economic modular design requiring low-capital investment, utilize existing spent fuel shipping casks for fuel transfer
to the modules, provide canistering of spent fuel, demonstrate licensability at a reactor site, and demonstrate that modular dry spent fuel storage can be accomplished independent of future shipping regulations.

CP&L, DOE, EPRI, and NUTECH Engineers are providing programmatic and financial support to the program.
Figure 1

PRIMARY COMPONENTS OF THE NUHOMS SYSTEM
Figure 3

ISOMETRIC VIEW OF MODULE ARRAY
SESSION 4 DISCUSSION PERIOD

"ECONOMICS OF DRY STORAGE" - WRITTEN CONTRIBUTION BY

D. DEACON (GEC-ESL)

The GEC-ESL cost estimates for various dry storage options of 500 teU and 1000 teU spent fuel capacity are shown in Tables 1 and 2.

### TABLE 1

TOTAL CAPITAL COSTS FOR AT-REACTOR STORAGE OF 1000 TEU OF SPENT FUEL (1983 DOLLARS)*

<table>
<thead>
<tr>
<th>Type of Storage</th>
<th>Total Cost for 1000 TEU of Storage (Dollars 000)</th>
<th>Unit Module Cost (Dollars 000)</th>
<th>No. of Modules</th>
<th>Cost of 1000 TEU (Dollars 000)</th>
<th>Initial Install Cost (Dollars 000)</th>
<th>Total Cost for 1000 TEU (Dollars 000)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Metal Casks</td>
<td>51,232</td>
<td>800</td>
<td>95</td>
<td>76,000</td>
<td>2,007</td>
<td>78,007</td>
</tr>
<tr>
<td>Drywells</td>
<td>34,312</td>
<td>23</td>
<td>2,331</td>
<td>53,613</td>
<td>5,129</td>
<td>58,742</td>
</tr>
<tr>
<td>Silos (Sealed)</td>
<td>29,417</td>
<td>83</td>
<td>583</td>
<td>48,389</td>
<td>3,845</td>
<td>52,234</td>
</tr>
<tr>
<td>Closed Cycle Vault</td>
<td>44,200</td>
<td>1,125</td>
<td>52</td>
<td>58,500</td>
<td>6,790</td>
<td>65,290</td>
</tr>
<tr>
<td>GEC MVDS</td>
<td>21,200</td>
<td>9,625</td>
<td>2</td>
<td>19,250</td>
<td>10,500</td>
<td>29,500</td>
</tr>
</tbody>
</table>

TABLE 2
TOTAL CAPITAL COSTS FOR AT-REACTOR STORAGE OF 500 TEU OF SPENT FUEL (1983 DOLLARS)*

<table>
<thead>
<tr>
<th>Type of Storage</th>
<th>Consolidated Fuel</th>
<th>Intact Fuel Assemblies</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Total Cost for 500 TEU of Storage (Dollars 000)</td>
<td>Unit Module Cost (Dollars 000)</td>
</tr>
<tr>
<td>Metal Casks</td>
<td>26,417</td>
<td>800</td>
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<tr>
<td>Drywells</td>
<td>19,262</td>
<td>23</td>
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<tr>
<td>Silos (Sealed)</td>
<td>15,517</td>
<td>83</td>
</tr>
<tr>
<td>Closed Cycle Vault</td>
<td>24,850</td>
<td>1,125</td>
</tr>
<tr>
<td>GEC MVDS</td>
<td>15,850</td>
<td>9,625</td>
</tr>
</tbody>
</table>

SESSION 5

DRY STORAGE - FUEL BEHAVIOUR (I)

Chairman:  S.J. Naqvi  
Ontario Hydro
LONG-TERM BEHAVIOUR OF IRRADIATED CANDU FUEL IN CONCRETE CANISTER STORAGE – TEST RESULTS

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ABSTRACT

Dry storage of irradiated fuel provides a long-term, viable and economical alternative to storage in water pools. In 1974, a Dry Storage Program was initiated at the Whiteshell Nuclear Research Establishment (WNRE) in Pinawa, Manitoba, Canada to demonstrate the concept. In the initial phases of the program, four demonstration canisters were constructed, two with electrical heaters and two for irradiated fuel storage. Since then, nine more concrete canisters have been constructed for irradiated fuel storage and experiments. To date, ≈1320 irradiated fuel bundles (≈18 000 kg U) are stored in concrete canisters at WNRE.

Three irradiated fuel storage experiments in concrete canisters are being conducted by Atomic Energy of Canada Limited (AECL) and Ontario Hydro (OH) in a joint program at WNRE. These experiments are known as the Easily Retrievable Basket (ERB) Experiment and Phases 1 and 2 of the Controlled Environment Experiment (CEX).
The ERB experiment involves the dry storage, at seasonally varying temperatures, of two irradiated natural UO\(_2\) Zircaloy-clad CANDU fuel bundles from the Pickering Nuclear Generating Station A, and 13 experimental, irradiated enriched UO\(_2\) and UC fuel bundles from the WWR-1 research reactor (WR-1). This experiment began in 1978. The bundles were examined before storage and after \(\approx 44\) months of dry storage at temperatures ranging from 55°C to -4°C. No degradation of the fuel or sheath was observed after the first storage period.

Both phases of the CEX experiment involve the storage of eight CANDU natural UO\(_2\) fuel bundles in air at 150°C. In Phase 1, the bundles are stored dry while in Phase 2 they are stored in air saturated with moisture. The main objective of these experiments is to demonstrate the long-term integrity of CANDU fuel sheathing in intact irradiated fuel elements and to examine the behaviour of intentionally defected irradiated fuel elements under various storage conditions.

Four bundles from each of the Pickering and Bruce Nuclear Generating Stations are being used in each of the CEX experiments. The Pickering fuel bundles contain 28 fuel elements while the Bruce fuel bundles contain 37 fuel elements. Two of each type of bundle were intentionally defected in each of the experiments to assess the behaviour of "intact" versus "defected" fuel elements. Phase 1 of the CEX experiment began in 1980 and Phase 2 in 1981. The fuel bundles in both experiments were retrieved in the spring of 1984 for their first interim examination. The bundles in Phase 1 were stored for \(\approx 41\) months in dry air at 150°C. Six of the bundles in Phase 2 were stored for \(\approx 30\) months and the remaining two were stored for \(\approx 24\) months in air saturated with moisture at 150°C.

The fuel bundles from the CEX experiments are currently being examined. No visual or dimensional change has been observed in the CEX-1 bundles examined to date. Axial gamma scanning also revealed no movement of radioisotopes. Metallographically and ceramographically, the sheath and fuel appeared unchanged from their pre-storage condition. In the one
element that was examined, some grain separation was observed along the fractures in the fuel pellet from the intentionally defected region. This change was not noted in any other fuel samples examined from this element.

Visual examination, gamma scans and end cap/sheath torque testing in the CEX-2 experiment showed no significant change from the pre-storage condition. One element showed minor corrosion of a bearing pad and corrosion product on the beryllium/zirconium alloy, used to braze the bearing pad to the sheath. Some localized pitting and corrosion product was also observed on the surface of an adjacent fuel element in the same bundle.
1. INTRODUCTION

A large quantity of irradiated CANDU* fuel (more than 300,000 bundles, or about 7,000 Mg U) is now stored in water-filled bays in Canada. Once the decay heat produced by the fuel bundles has reached a sufficiently low level (approximately 0.5 W/kg U), interim storage may be continued either under water or in a dry storage facility. Most fuel storage information in Canada is based on wet storage (Walker 1979; Remington 1983). However, interim dry storage, in either convection vaults (Boulton, 1978; Mayman, et al., 1976) or concrete canisters (Boulton, 1978; Freire-Canosa, 1982, 1984; Naqvi 1982; Ohta, 1978), has potential advantages in terms of ease of operation, little maintenance, no generation of secondary wastes, safety, safeguards and decommissioning (Rummery and McLean, 1979). Consequently, a dry storage program involving concrete canisters was initiated in Canada.

In this paper, we will describe the Concrete Canister Program at WNRE, and the joint AECL/Ontario Hydro Experimental Test Program in concrete canisters, and give the results of the first interim examination of irradiated fuel bundles after dry storage for many months.

2. BACKGROUND ON THE AECL CONCRETE CANISTER PROGRAM

In the spring of 1974, WNRE began a development and demonstration program on the concrete canister dry storage concept. This was undertaken as a possible alternative to water pools for the interim storage of irradiated fuel on the recommendation of the Committee for Assessing Fuel Storage (Morgan, 1977). During 1974, two cylindrical canisters were constructed. One was electrically heated and the other was designed for

* Canadian Deuterium Uranium, Canada’s heavy-water moderated and cooled, natural-uranium fuelled power reactor. The present CANDU fuel bundles are fabricated from Zircaloy-4 tubing and are fuelled with natural UO₂ pellets.
irradiated-fuel storage. Figure 1 shows the fuelled, cylindrical demonstration canister. The following year, two square canisters were constructed. Again, one was electrically heated and the other was designed for irradiated-fuel storage. Figure 2 shows the fuelled, square demonstration canister. The electrically heated canisters were constructed primarily to demonstrate the basic concrete canister concept, and to assess their structural integrity, and heat transfer and shielding capabilities, under various weather conditions.

In 1975 and 1976 two of the demonstration canisters were loaded with irradiated enriched \( \text{UO}_2 \) from the Whiteshell Nuclear Research Reactor, WR-1 and natural \( \text{UO}_2 \) from Ontario Hydro's Douglas Point Nuclear Generating Station. Since the condition of the irradiated fuel sheaths was unknown and the concrete canister concept was unproven, the welded carbon steel baskets containing the fuel bundles were filled with an inert atmosphere to prevent oxidation of the \( \text{UO}_2 \) in case any fuel failures existed. The canister lid was welded to the internal steel canister liner to provide another containment barrier. Also, the maximum temperature on the hottest bundle in the demonstration canisters was limited to 250°C by only loading fuel that had cooled for a sufficiently long time. Additional details on the concrete canister program may be obtained from Ohta (1978) and Sochaski (1984).

Since the construction of the electrically heated and fuelled demonstration concrete canisters, nine additional canisters have been constructed at WNRE. Three are used exclusively for experiments, one is used jointly for an experiment and the storage of WR-1 irradiated fuel, and the remainder are used solely for WR-1 irradiated fuel storage. To date, ~1320 fuel bundles (~18 000 kg U) are being stored in concrete canisters at WNRE. Figure 3 is an aerial view of the WNRE concrete canister storage site.
3. JOINT AECL - ONTARIO HYDRO EXPERIMENTAL PROGRAM

AECL and Ontario Hydro are participating jointly in three irradiated-fuel storage experiments in concrete canisters at WNRE. The three experiments are known as the Easily Retrievable Basket (ERB) Experiment (dry storage at seasonally varying temperatures), and the Controlled Environment Experiments (CEX), Phase 1 (dry storage in air at 150°C) and Phase 2 (storage in air saturated with moisture at 150°C).

The ERB experiment is conducted at seasonally varying temperatures since these conditions are representative of the temperature fluctuations experienced by the remainder of the irradiated-fuel bundles in the storage canisters. However, in a large, interim dry storage facility, the fuel temperature is expected to be initially as high as 150°C, due to the decay heat output of the fuel bundles. Consequently, a storage temperature typical of an interim storage vault was chosen for the CEX experiments.

The fuel bundles in the ERB experiment were loaded into a concrete canister in 1978 October. The CEX-1 canister was loaded in 1980 October, and the initial loading of the CEX-2 canister was in 1981 November, with a final loading in 1982 June. All of the CANDU fuel bundles in the above experiments were characterized before loading into the canisters.

The primary objective of the experimental program is to demonstrate the long-term integrity of irradiated CANDU fuel bundles, with intact and defected fuel elements, under dry storage conditions. Secondary objectives are as follows:

(1) Compare the effects of storage on intact and defected irradiated-fuel elements at 150°C in both dry air and air saturated with moisture.
(2) Study the effects (if any) of fission products deposited on the inner sheath surface on long-term sheath integrity. In particular, possible sheath/fission-product interactions will be investigated with and without exposure to dry and moist air at 150°C.

(3) Monitor the quantity and migration of specific radionuclides in the fuel elements under long-term dry storage at 150°C.

(4) Study the effects, if any, of both dry and moist air at 150°C on the rate of release of radionuclides from intentionally defected fuel elements.

(5) Assess the effect of an internal CANLUB* coating on fuel sheaths on each of the above objectives.

(6) Study the long-term resistance of the sheath to external corrosion.

3.1 EASILY RETRIEVABLE BASKET EXPERIMENT

The objective of the ERB experiment is to assess the storage behaviour of irradiated-fuel bundles in dry air at seasonally varying temperatures. The ERB is unlike the standard canister basket. The lid on the ERB is bolted onto the basket whereas the lids on the standard baskets are welded on. The removable lid enables easy retrievability and inspection of the fuel bundles. The ERB is also instrumented with thermocouples and backfilled with air, whereas the standard baskets are not instrumented and are backfilled with helium.

Two typical CANDU, natural UO_2 fuel bundles from the Pickering Nuclear Generating Station-A (PNGS-A) and 13 experimental, enriched UO_2 and UC fuel bundles from WR-1 were loaded into the canister in 1978 October (Oldaker et al., 1979). Figure 4 is a schematic of a Pickering fuel

* CANLUB is graphite coating (i.e. lubricant) up to 20 µm thick, applied to the inner surface of the sheath of the outer elements in each bundle.
bundle. The two Pickering bundles were discharged from the reactor in 1975 December. They were irradiated at a peak outer-element linear rating of 43 kW/m and reached an average outer-element burnup of approximately 190 MWh/kg U.

3.2 CONTROLLED ENVIRONMENT EXPERIMENT - PHASE 1

This experiment involves the storage, in dry air at 150°C, of four irradiated CANDU fuel bundles from the Bruce Nuclear Generating Station-A (BNGS-A) and four irradiated CANDU fuel bundles from the Pickering Nuclear Generating Station-A (Oldaker, 1979). Figure 5 is a schematic of a Bruce fuel bundle. Two of the Bruce fuel bundles were of the CANLUB plenum design while the other two were of the CANLUB non-plenum design. (A gas plenum was included in the earlier Bruce fuel bundles to accommodate the release of fission gas; the most recent design does not include a plenum). Two of the Pickering bundles were of the non-CANLUB type whereas the remaining two Pickering bundles were of the CANLUB type.

CANLUB and non-CANLUB bundles were chosen for this experiment to investigate the effect of CANLUB graphite on fuel behaviour during long-term storage. CANLUB graphite adsorbs many active chemical species, and its presence in operating fuel elements may limit the access of stress corrodants, such as I, Cs and Cd, to the sheath. Iodine is known to promote stress-corrosion cracking (SCC) (Wood, 1972) in zirconium alloys and, more recently, it has been discovered that Cs and Cd cause metal vapour embrittlement (MVE) (Hunt et al., 1981) in these zirconium alloys.

All the outer elements (except three which were characterized) in one of each type of fuel bundle were deliberately defected by drilling a 3-mm-diameter hole through the sheath before storage, to assess the storage behaviour of intact versus intentionally defected fuel elements. The fuel bundles used in this experiment were discharged from the reactor between 1973 and 1977. They were irradiated at peak outer-element linear ratings ranging from 33 kW/m to 50 kW/m, and reached average outer-element burnups ranging from 186 MWh/kg U to 222 MWh/kg U. The bundle types were chosen to represent the inventory of discharged CANDU fuel bundles.
3.3 CONTROLLED ENVIRONMENT EXPERIMENT – PHASE 2

This experiment also involves the storage of four Bruce and four Pickering irradiated fuel bundles, but this time in air saturated with moisture at 150°C (Walker and Cracknell, 1981). A moist-air environment is being used to simulate a storage situation that involves the transfer of water from the water pools to the containers in which the bundles will be stored, via the bundle and/or bundle container surfaces, or in defected elements. High- and low-flux fuel bundles were chosen for this experiment to determine the effect of fission products on fuel integrity during storage, since fission-product gas release is most sensitive to fuel power (temperature) and burnup (Notley and Hastings, 1980). All four of the Bruce bundles were of the CANLUB variety. Two of the Pickering fuel bundles were of the CANLUB variety while the remaining two were of the non-CANLUB type. The fuel bundles in Phase 2 of this experiment were discharged from the reactor between 1973 and 1980. They were irradiated at peak outer-element linear ratings ranging from 22 kW/m to 53 kW/m and reached average outer-element burnups ranging from 185 MWh/kg U to 262 MWh/kg U. As in Phase 1, all of the outer elements (except three which were characterized) in one of each type of bundle were deliberately defected before storage.

4. PRE-STORAGE CHARACTERIZATION

The pre-storage characterization has been described previously (Wasywich et al., 1982). To review, the characterization began with a preliminary visual inspection, photographing of the fuel bundles through the hot-cell periscope, and measuring the dimensions of the fuel elements using the Diaprof (Swanson et al., 1980), an automated apparatus for measuring fuel-element diameters and profiles before element removal from a fuel bundle.

Then, specific outer-ring elements were removed from each of the bundles for a more detailed examination. A non-destructive examination was
performed on one outer-ring element from each bundle, which became a "control" element, while a destructive examination was performed on another outer-ring element from each bundle. Following examination, the control elements were loaded into the canisters with the remainder of the fuel bundles. Since it was not practical to perform a detailed examination on all the elements in each fuel bundle, a control element provides a means of monitoring the probable behaviour of the remaining elements in the bundle during storage. The examination performed on the control elements consisted of element diameter and profile measurements using a Profilometer, end-plate/end-cap torque tests, stereophotography and axial gamma-ray scanning for total activity and cesium fission products.

The destructive examination included the examination performed on the control elements plus the following:

(1) Fission-gas analysis and internal free-void volume measurements.

(2) Ring tensile tests.

(3) Hydrogen/deuterium analysis of the Zircaloy components.

(4) Metallography of the sheath.

(5) Ceramography of the fuel.

(6) Radiochemical analysis of cores removed from cross-sections of fuel.

(7) Radial gamma-ray scanning of the fuel.

(8) Burnup analysis of the fuel.
5. RESULTS OF THE FIRST INTERIM EXAMINATION

5.1 BACKGROUND

The bundles in the ERB experiment were retrieved from the concrete canister at WNPE on 1982 July 6 for their first interim examination. The bundles had been stored dry for 44 months in air at seasonally varying temperatures. The fuel bundles were reloaded into the canister on 1982 July 29 for continued storage, following an examination of the bundles and the removal of specific fuel elements for a more detailed examination. The next retrieval is tentatively scheduled for 1992.

During the first storage period, the bundles experienced storage temperatures ranging from a high of 55°C shortly after they were loaded into the canister to a low of -4°C in 1982 January. Since then, the lowest storage temperature the bundles experienced was -13°C, in 1984 January.

The bundles in the CEX-1 experiment (dry storage at 150°C) were retrieved from the concrete canister on 1984 March 29, after 41 months of storage. They were reloaded into the canister on 1984 May 28 for continued storage. The bundles were examined and specific elements were retained for a detailed examination. So far, a detailed destructive examination has been performed on only one of the intentionally defected elements removed from one of the higher burnup Pickering bundles. The results of that examination will be discussed later in this section. The timing and scope of the next retrieval of the bundles in this experiment will be dictated by the results of the first interim examination.

The bundles in the CEX-2 experiment (storage in air saturated with moisture at 150°C) were retrieved from the canister on 1984 June 1. Two of the CEX-2 Pickering bundles were stored for ~24 months while the remaining six bundles were stored in the canister for ~30 months. The bundles from this experiment are currently being examined in the hot-cell facilities at WNRE. They will be returned to the canister for continued storage following their examination and the removal of specific elements for detailed
examination. Arrangements are currently being made to include in the CEX-2 experiment during the next storage period two well-characterized CANDU fuel elements that have defected in-reactor. Naturally defected fuel elements are being included in the program since their storage behaviour may differ from intentionally defected elements, because of the deterioration the naturally defected elements may have experienced during irradiation. The next interim examination is currently scheduled for 1986.

Results from the first interim examination performed on the fuel bundles involved in the above experiments are presented below.

5.2 VISUAL EXAMINATION

All of the bundles were examined visually using the hot-cell periscope and stereomicroscope. No change was observed visually in any of the fuel elements in both of the ERB and CEX-1 dry experiments. Figure 6 shows the typical condition of the sheaths in the hot, dry, CEX-1 experiment. Also shown in Figure 6 are the 3-mm-diameter holes (intentional defects) that were drilled in the elements.

A slight change was observed visually in a small number of the elements in the CEX-2 moist experiment. For example, white stains were observed on a small number of elements, as shown in Figure 7. A stain and associated pitting were observed on another element (see Figure 8) and corrosion products were seen on the Be/Zr sheath/bearing-pad braze alloy on a second element (see Figure 9). The pitting on the sheath and corrosion along the bearing pad were isolated cases and not representative of the general condition of the bundles. The scanning electron microscope and energy dispersive X-ray analysis of the corrosion products revealed Fe, Ni, Cr and Zr.

5.3 ELEMENT DIMENSIONAL MEASUREMENTS

Element diameters were measured before and after the first storage period. So far, no change in element diameter has been observed on any of the elements. Typical pre- and post-storage element diameter profilometer
traces of a Pickering control element from the hot, dry experiment are shown in Figures 10 and 11, respectively. Typical pre- and post-storage average element diameters were $15.27 \pm 0.01$ mm over the mid-pellet region in the above control element and $15.38 \pm 0.01$ mm over the pellet/pellet interface. The above element was irradiated at a peak outer-element linear rating of $48$ kW/m and reached an average burnup of $222$ MWh/kg U. Figure 12 shows a post-storage diameter trace of an intentionally defected Pickering element from the CEX-1 experiment. As shown in the trace, very little difference in element diameter was observed in the vicinity of the drilled hole and away from that region.

5.4 **FISSION-GAS ANALYSIS**

Two fuel elements were punctured for fission-gas analysis after 44 months of dry storage in the ERB experiment. (Fission-gas analysis was not performed in the ERB experiment before storage). The burnups of the two elements that were punctured varied from 185 to 192 MWh/kg U, and the peak outer-element linear power ratings the bundles experienced varied from 48 to 50 kW/m. Fission-gas analysis is planned for both CEX experiments, but has not been performed yet. The post-storage gas releases ($\text{Xe} + \text{Kr}$) in the ERB experiment varied from 0.7% to 1.4% in the elements that were punctured. The measured post-storage gas releases were typical of those expected from Pickering CANDU fuel elements irradiated under the above conditions. The analysis confirmed the integrity of the ERB fuel elements after $\frac{1}{4}$ years of dry storage in air.

5.5 **VOID VOLUME MEASUREMENTS**

Knowledge of the free-void volume within the fuel elements is useful in calculating the internal gas pressure and sheath stresses in intact fuel elements at various storage temperatures. Consequently, the free-void volume within a number of CEX fuel elements was measured during the pre-storage examination. It was decided to also perform these measurements on specific fuel elements from all three experiments during the first interim examination. So far, the post-storage measurements have been performed only on Pickering elements from the ERB experiment.
The void volume measurements were performed after the fuel elements had been punctured for fission-gas analysis. Following that analysis, the puncture hole was sealed with an epoxy resin. After the epoxy resin had cured, the elements were repunctured, the air was evacuated from them and its volume measured.

The pre-storage free-void volumes in the Pickering fuel elements used in the CEX experiments ranged from 2.4 to 2.8 mL at STP. At a storage temperature of 150°C, these volumes would have resulted in internal gas pressures of less than 0.5 MPa and sheath hoop stresses of approximately 8.5 MPa. It is unlikely that environmentally induced sheath failures would occur under these conditions (Hunt et al. 1981).

The free-void volumes of Pickering fuel elements used in the ERB experiment were measured during the first interim examination and ranged from 2.5 to 3.0 mL at STP. At the peak storage temperature of 55°C that the ERB bundles experienced, these volumes would have produced internal gas pressures of less than 0.2 MPa and sheath hoop stresses of about 3.4 MPa. The above stress levels would not likely be sufficient to cause sheath defects during storage.

5.6 END-PLATE/END-CAP TORQUE TESTS

Torque tests were performed on specific fuel elements in the CEX experiments before storage, to evaluate the end-plate/end-cap weld strength. Similar tests were conducted on specific fuel elements from all three experiments after the first storage period, to monitor the effects of storage on end-plate/end-cap weld strength and, in turn, bundle integrity. This will permit an evaluation of the ability of irradiated bundles to withstand handling during storage and transportation operations.

To perform the torque test, the fuel element is held rigidly in a fixture (typically a three-jaw chuck of a lathe). The section of attached end plate is then rotated about the element’s longitudinal axis, using a torque wrench equipped with a maximum torque-indicating dial, until the end plate breaks free from the end cap. The maximum torque read on the wrench
indicator is a representative measure of the weld strength. No significant difference was observed between the pre- and post-storage results. In the ERB experiments, the post-storage peak and average torque values required to fail the end-plate/end-cap welds in the two Pickering bundles were 13.6 Nm and 11.9 Nm, respectively. This test was not performed on any of the ERB elements before storage. However, very little difference was observed between the post-storage ERB results and the pre-storage CEX-2 results (peak and average torque values of 15.3 Nm and 11.7 Nm, respectively) for a bundle with a similar irradiation history.

In the CEX-1 experiment, this test was performed on only two fuel elements from one of the Pickering bundles before storage, since end-plate/end-cap weld failures were observed in that bundle when it arrived at WNRE. The average pre- and post-storage torque values required to fail the end-plate/end-cap welds in the above bundle were 8.8 Nm and 8.5 Nm, respectively.

Pre-storage end-plate/end-cap torque tests were performed on all of the CEX-2 bundles; however, so far, the post-storage tests have only been performed on the Bruce fuel bundles. Average pre- and post-storage torques required to fail the Bruce end-plate/end-cap welds were 9.6 Nm and 9.9 Nm, respectively.

The good agreement between the pre- and post-storage torque test results in all of the experiments indicates that no degradation of the end-plate/end-cap welds occurred during the first storage period.

5.7 RING TENSILE TESTS

These tests were performed on 10 mm-wide sections of sheaths removed before storage from specific fuel elements in each of the bundles in all three experiments. So far, post-storage tests have only been performed on sheaths from the ERB experiment. The tests were conducted at room temperature using a servohydraulic tensile testing machine.
Fairly good agreement was observed between the pre- and post-storage ring tensile test results in the ERB experiment, suggesting there was essentially no change in mechanical properties during storage. Typical average Ultimate Tensile Strength (UTS) values for sheath material before and after storage were 1014 MPa and 975 MPa, respectively. Typical average uniform elongation values before and after storage were 4.4% and 5.2%, respectively.

5.8 HYDROGEN/DEUTERIUM ANALYSES

Since hydrogen and deuterium concentrations in the Zircaloy components of a fuel bundle can have a significant effect on the mechanical properties and, consequently, the integrity of the fuel bundle during transportation and storage, the pre-storage concentrations were measured.

Hydrogen and deuterium analyses were performed before storage on the end plates, end caps and fuel sheaths from specific fuel elements in each of the experiments. A hot vacuum extraction and mass spectrometry procedure was used.

So far, post-storage analysis has been performed on all the Zircaloy components in the Pickering bundles from the ERB experiment. The majority of the results generated before and after the first storage period agree within the precision of the measurements, indicating that no movement of hydrogen/deuterium has occurred during the first storage period. Typical average hydrogen equivalent concentrations in the sheaths before and after the first storage period were 59 ± 8 μg/g and 70 ± 9 μg/g, respectively. Typical average hydrogen equivalent concentrations in the end caps before and after the first storage period were 72 ± 8 μg/g and 50 ± 7 μg/g, respectively. The end-plate samples in the ERB experiment were not analyzed before storage, but they were in the CEX experiments. So far, only the

* Hydrogen equivalent concentration (μg/g) = Hydrogen concentration (μg/g) + 1/2 deuterium concentration (μg/g).
end plates from the ERB and CEX-1 experiments have been analyzed for hydrogen and deuterium after the first storage period. The typical pre-storage average hydrogen equivalent concentration was $49 \pm 7 \ \mu g/g$ in the end plates from the CEX-2 Pickering bundles that had experienced irradiation conditions similar to the Pickering bundles in the ERB experiment. The post-storage average hydrogen equivalent concentration was $52 \pm 7 \ \mu g/g$ in the end plates from the Pickering bundles in the ERB experiment. In CEX-1, the pre-storage hydrogen/deuterium concentrations in the end plates was measured in only one of the Pickering bundles. The pre- and post-storage hydrogen equivalent concentrations were $62 \pm 8 \ \mu g/g$ and $70 \pm 8 \ \mu g/g$, respectively, in the end plates of that bundle.

5.9 METALLOGRAPHIC AND CERAMOGRAPHIC EXAMINATION

Sections of sheathing and fuel from the ERB and CEX-1 experiments were examined in detail following the first storage period, to determine if they had experienced any degradation due to the storage conditions. No degradation was observed in the ERB experiment during the above examination. Metallographically and ceramographically, the sheath and fuel in the CEX-1 experiment appeared unchanged from their pre-storage condition. However, more UO$_2$ grain "pull-out" was observed along the fractures in the fuel pellet located in the intentionally defected region than in the fuel away from that region in the CEX-1 experiment. At the present time, the cause of the pull-out is uncertain.

Photomacrographs of longitudinal sections of fuel and sheathing taken from an intentionally defected CEX-1 bundle before and after the first storage period are shown in Figures 13 and 14, respectively. Figure 14 was obtained $\approx 300$ mm from the intentional defect. As illustrated in the photographs, there is no apparent difference in the appearance of the fuel. Figure 15 is a photomacrograph of a longitudinal section of fuel and sheathing taken from the above bundle in the intentionally defected region following the first storage period. In Figure 15, the cracks in the fuel are not as well defined as they were before storage, or away from the defected region after storage. Photomicrographs of the cracks in the fuel away ($\approx 300$ mm) from and near to the defected region are shown in Figures 16
and 17, respectively. More grain pull-out was observed along the fractured fuel surfaces near the intentional defect than away from it.

The zirconium oxide layer on the outer sheath surface, the zirconium hydride/deuteride distribution in the sheath, and the condition of the inner sheath surface were also examined. No changes were observed in these areas in either of the CEX-1 and 2 experiments after the first storage period.

5.10 AXIAL GAMMA-RAY SCANNING

Fuel elements from the ERB and CEX-1 experiment were gamma-scanned to determine the fission-product distribution following the first storage period. Work on CEX-2 fuel elements is in progress. In the ERB experiment, outer-ring elements were gamma-scanned before and after the first storage period. The fission-product distributions in the two cases were almost identical, indicating no evidence of fission-product migration. In CEX-1, control elements were gamma-scanned before and after the first storage period. No detectable difference was observed between the pre- and post-storage axial fission-product distributions, as shown in Figure 18. The CEX-2 elements that have been gamma-scanned so far also indicate no fission-product migration during the first storage period.

5.11 RADIAL GAMMA-RAY SCANNING

In the ERB experiment, cross-sections of fuel from outer Pickering fuel elements were obtained after the first storage period, to determine the radial distribution of fission products. Work on the ERB experiment is complete and the CEX-2 analytical work is in progress. The pre-storage and post-storage scans from the ERB experiment are very similar (see Figure 19), indicating no significant radial movement of fission products during storage.
5.12 ANALYSIS OF CEX-2 SOLUTIONS

To saturate the air with moisture in the CEX-2 experiment, 100 mL of distilled water was placed in each fuel bundle container before storage. This water was removed following the first storage period and is being analyzed. Preliminary gamma-ray spectrometry data show a wide range in the specific activity of $^{137}$Cs. The $^{137}$Cs activity ranges from $1.77 \times 10^6$ to $6.27 \times 10^7$ Bq/mL for the eight sample solutions. The $^{134}$Cs/$^{137}$Cs ratio was generally greater than 0.8, which is higher than that observed in the fuel, where this ratio is typically less than 0.7. These results suggest that the fission-product activity found in the solutions originates mainly from crud deposits on the surfaces of the fuel bundles and not from the fuel matrix. The deposits may have been picked up by the fuel bundles in the hot cells, the water storage bays, or the reactor.

6. SUMMARY AND CONCLUSIONS

The results of the first interim examination of irradiated CANDU fuel stored in concrete canisters are summarized below.

(1) The fuel bundles in the ERB experiment experienced no apparent change after 44 months of dry storage in air at seasonally varying temperatures.

(2) In the CEX-1 experiment, no significant degradation of the fuel or sheaths was observed after dry storage in air at 150°C for approximately 3 1/2 years.

(3) In the CEX-2 experiment, some corrosion was observed on a bearing pad and some corrosion product was observed on the associated Be/Zr sheath/bearing-pad braze alloy as well as on the sheath of an adjacent element. Since these were isolated cases, the corrosion may be associated with contaminants.
(4) Fuel-element diameters were measured in all experiments, before and after storage. No change in element diameters was observed.

(5) Axial gamma-ray scans on fuel elements from all three experiments before and after storage and radial scans on fuel from the ERB experiment before and after the first storage period showed no evidence of fission-product migration.

(6) The measured, post-storage, fission-gas (Xe + Kr) release for the Pickering fuel elements in the ERB experiment varied from 0.7% to 1.4%. These results are typical of releases expected from such elements. Post-storage fission-gas measurements have not yet been performed for the CEX experiments.

(7) Internal gas pressures and sheath hoop stresses were derived from free-void volume measurements performed on a number of elements from each of the experiments. The results indicated that the internal gas pressures and resulting sheath hoop stresses were insufficient to cause environmentally induced sheath failures during dry storage at temperatures of up to 150°C.

(8) The majority of the hydrogen/deuterium results generated before and after the first storage period in the ERB and CEX-1 experiments agreed within the precision of the measurements, indicating that no hydrogen movement had occurred during the first storage period.

(9) No significant difference was observed between the results of the pre- and post-storage end-plate/end-cap torque tests, which suggests that the mechanical properties of the end-plate/end-cap welds did not change during the first storage period in the ERB and CEX-1 and 2 experiments.

(10) Fairly good agreement was observed between the pre- and post-storage ring tensile test results for fuel sheaths in the ERB experiment, suggesting that the mechanical properties of the
sheaths were not affected by the storage conditions. Post-storage tests have not yet been performed for the CEX experiments.

(11) The zirconium oxide layer on the outer sheath surface, the zirconium hydride/deuteride distribution in the sheath, and the condition of the inner sheath surface were also examined in specific elements from the ERB and CEX-1 experiments. No changes were observed in these areas after the first storage period.

(12) The water solutions used in the CEX-2 experiment are being analyzed to determine if any fission-product leaching from the defective elements has occurred. Preliminary results suggest that the fission-product activity found in the solutions originates mainly from crud deposits on the surfaces of the fuel bundles and not from the fuel matrix.

The above results are from the post-storage examination of specific fuel elements from the ERB experiment, the first interim preliminary examination of the CEX-1 and 2 fuel bundles, and the first interim detailed examination of one CEX-1 fuel element. The results may not be entirely representative of the remaining fuel elements that are yet to be examined, since a wide variety of fuel bundles was chosen for these experiments.

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REFERENCES


FIGURE 1
Cylindrical Demonstration Canister Containing 138 WR-1 UO₂ Fuel Bundles
FIGURE 2
Square Prismatic Demonstration Canister Containing 360 Douglas Point UO₂ Fuel Bundles
FIGURE 4
Pickering Fuel Bundle Design

BUNDLE ASSEMBLY NOTE
1. SPACING BETWEEN ADJACENT SHLATH SURFACES IS 1.0 mm MINIMUM

*IRRADIATED FUEL DIMENSIONS
ALL DIMENSIONS IN MILLIMETERS.
FIGURE 5
Bruce Fuel Bundle Design
FIGURE 6
General Condition of Intentionally Defected Pickering Fuel Elements After 41 Months of Dry Storage in Air at 150°C. (Typical of the Post-Storage Conditions of the Fuel Elements in the Dry Experiment).

FIGURE 7
General Condition of Bruce Fuel Elements After Approximately 30 Months of Moist Storage in Air at 150°C
FIGURE 8
Pitting Observed on the Outer Sheath Surface of a Bruce Fuel Element
After Approximately 30 Months of Storage in Moist Air at 150°C

FIGURE 9
Corrosion of the Beryllium/Zirconium Braze Alloy
Observed Alongside a Bearing Pad on a Bruce Fuel Element
After Approximately 30 Months of Storage in Moist Air at 150°C
FIGURE 10
Pre-Storage Diameter Trace of Element 1
From CEX-1 Pickering Bundle 26244C
FIGURE 11
Diameter Trace of Element 1 From CEX-1 Pickering Bundle 26244C
After 41 Months of Dry Storage in Air at 150°C
FIGURE 12
Diameter Trace of Intentionally Defected Element 3
From CEX-1 Pickering Bundle 26244C
After 41 Months of Dry Storage in Air at 150°C
FIGURE 13
Pre-Storage Condition of the UO₂ Fuel From Element 3 CEX-1 Pickering Bundle 26244C

FIGURE 14
Condition of the UO₂ Fuel Approximately 300 mm From the Intentional Defect in Element 3 CEX-1 Pickering Bundle 26244C After 41 Months of Dry Storage in Air at 150°C
FIGURE 15
Condition of the UO₂ Fuel Adjacent to the Intentional Defect
in Element 3 CEX-1 Pickering Bundle 26244C
After 41 Months of Dry Storage in Air at 150°C
After 41 Months of Dry Storage in Air at 150°C

**FIGURE 16**
Condition of the UO₂ and Grain Pull-Out Along a Fracture in the Fuel Approximately 5.0 mm From the Intentional Defect in Element 3 CEX-1 Pickering Bundle 26244C
After 41 Months of Dry Storage in Air at 150°C

**FIGURE 17**
Condition of the UO₂ and Grain Pull-Out Along a Fracture in the Fuel Adjacent to the Intentional Defect in Element 3 CEX-1 Pickering Bundle 26244C
After 41 Months of Dry Storage in Air at 150°C
FIGURE 18
Pre- and Post-Storage Axial Gamma Scans of the Control Element From CEX-1 Bundle E04313
FIGURE 19
Pre- and Post-Storage Radial Gamma Scans of an Element From the ERB Experiment
FUEL ROD AND CRUD BEHAVIOUR UNDER LONG-TERM DRY FUEL STORAGE CONDITIONS

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ABSTRACT

A testing program using eight commercial PWR and BWR spent fuel rods was conducted to investigate the rods' long-term stability under a variety of possible dry storage conditions. The objective of this project is to provide the NRC with the information to confirm or establish spent fuel, dry storage licensing positions for long-term, low temperature (<523 K) spent fuel rod behavior during dry storage and for radioactive contamination arising from spallation of cladding crud. The two-year test was comprised of three interim nondestructive examinations. This paper presents the results of the third examination conducted to determine any degradation in all eight fuel rods after being subjected to 13,168 h at temperatures between 502 K and 490 K. During this examination, visual observations, diametrical measurements, and isotopic analysis of smears were used to assess the fuel rod behavior and particulate release. The PWR fuel rods showed no measurable change from the pretest condition. The artificial defects had not changed and no diametrical growth in the cladding occurred. A BWR fuel rod replaced one that breached after the second interim examination, and a 1.9 cm crack developed in this rod at the bottom defect. About 17% cladding deformation was observed at the defect. Minute amounts of crud consisting of the cobalt-60 isotope fell into the fuel rod capsules. The BWR fuel rod removed after the second interim examination was destructively examined. The fuel was oxidized to U₃O₈ in the area of the cladding defect. The extensive fuel oxidation may result from diffusion of oxygen through the open porosity. Release of fission gases in a sealed fuel rod capsule may also be attributed to fuel oxidation along open grain boundaries.

The results of the testing with the BWR fuel would indicate that dry storage above 490 K to 502 K is not acceptable for this particular fuel used. This fuel may be atypical of current fuel design and fabrication techniques. The BWR results may not be applicable to PWR fuel because of the wide differences observed between PWR and BWR fuel rod behavior. The PWR fuel rods are being destructively examined to determine the extent of oxidation, but additional effort is required to determine oxidation mechanisms and identify the causes of the differences in behavior between PWR and BWR UO₂ fuel.
1. INTRODUCTION

The contamination potential of spent fuel during long-term, low-temperature (<523 K) dry storage depends on the fuel rod's performance in the atmosphere selected for storage. Rods that have been breached in the reactor are not routinely isolated; therefore, some rods with cladding perforations may be stored in dry environments. Contamination may result from both spallation of the crud coating or fuel particulate and fission gas releases from a perforated rod which fails during storage.

A long-term, 8-fuel-rod test using commercial fuel was initiated at 502 K covering a wide range of storage atmospheres, rod types, and cladding conditions. This test was part of a long-range project to evaluate the behavior of spent fuel during dry fuel storage conditions. Results from this project provide the Nuclear Regulatory Commission (NRC) with information to confirm or establish spent fuel, dry storage licensing positions regarding long-term, low-temperature (<523 K) spent fuel rod behavior during dry storage, and radioactive contamination arising from spallation of cladding crud.

In an unlimited air atmosphere, oxidation of UO₂ may occur with a concurrent volume expansion and rupture of the cladding. The contamination potential may be enhanced by: (a) oxidation of the fuel along the grain boundaries which would release fission gases trapped in the grain boundaries, (b) fallout of fuel particulate from the rupture, and (c) spallation of the crud from stresses imposed on the cladding by fuel expansion. Similar behavior, although at different rates, may occur with other atmospheres containing impurities such as an inert atmosphere with moisture or some other oxidant. Estimates have been made of expected maximum storage temperatures, (Blackburn, 1978) but information is needed to assess a satisfactory storage temperature with regards to defected rods in an oxidizing environment.
Four intact and four defected rods were tested. The four defected rods were examined during the first and second nondestructive interim examinations (Einziger and Cook, 1983; Einziger and Cook, 1984). The first interim examination was conducted after a 2,235 h exposure at temperature, and a second interim examination was conducted after a total of 5,962 h at temperature. The third and final nondestructive examination was conducted after 13,168 h, and a breached rod from the second interim examination, replaced with another fuel rod, was destructively examined. This paper presents both the results of the third interim nondestructive examination and a summary of the destructive examination of the breached fuel rod.

2. EXPERIMENTAL PROCEDURE

Four PWR fuel rods from the H.B. Robinson reactor and four BWR rods from the Peach Bottom reactor were heated in a furnace to simulate temperature occurring during dry storage conditions. These fuel rods were described previously (Einziger and Fish, 1982; Einziger and Cook, 1984). Four rods (two PWR and two BWR) each contained artificial defects in the form of 0.76 mm diameter holes placed at different orientations and axial positions (Table 1). Stainless steel capsules with an inside diameter of 1.75 cm were used to contain the fuel rods with each one in its own atmosphere (Table 2). One defective rod of each type was placed in a sealed capsule containing 0.1 MPa of an argon-1% He mixture. The other two defected rods were placed in capsules which terminated at each end with a series of 2 μm and 15 μm in-line filters open to the cell atmosphere. These filter sizes were based on fuel particle sizes expected from ruptured fuel rods (Lorenz et al, 1980). The intact rods were handled in a similar fashion. One intact rod of each type was placed in a sealed capsule containing 0.1 MPa of an argon-1% He mixture. Also one intact rod of each type was placed in a capsule containing argon-1% He mixture. The leak rate on the sealed capsules was a minimum of $2.5 \times 10^{-7}$ standard cm$^3$/s.
The fuel rods were heated in a shielded 14 zone, 12.8 m long clamshell furnace capable of holding the 8 encapsulated, unmodified LWR fuel rods (Figure 1). The fuel rod capsules were placed around an instrument train which contained 10 axially located thermocouples (Figure 2), and furnace-control thermocouples indicated a +3 K radial temperature gradient, with the center of the furnace being the hottest. The axial variation of the hot zone temperature was +2 K with a time variation of +2 K.

The furnace temperatures were read and printed on paper tape once an hour with a Fluke data logger, but during the least 10 weeks prior to shutting the furnace down for the final nondestructive examination, the fluke data logger was connected to Apple II+ personal computer for storing the temperature data on floppy disks for subsequent data reduction.

For each furnace campaign, the furnace was brought to temperature over a 12 h period and, other than for power outages, ran continuously until the interim examinations (Figure 3). The furnace was allowed to cool by natural means at a rate less than 5 K/h. The furnace was operated at 502 K for 5,932 h and then gradually decreased to 490 K during the next 7,206 h (Table 3).

Peach Bottom rod PH462-E3 was removed from the furnace after 5,962 h at 502 K. This rod was replaced with PH462-C5 which was heated for 7,206 h at temperatures decreasing from 502 K to 490 K. The remaining 7 rods were heated for 13,168 h, 5,962 h at 502 K and 7,206 h from 502 to 490 K.

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The fuel rod capsules were removed from the furnace on July 13, 1984. Gas samples were taken from the sealed capsules, and then the fuel rods were removed from the capsules. The capsules were swabbed to check for any loose crud or fuel particulate. The filters and swabs were weighed to determine the amount of material picked up from the capsules. These swabs were gamma-scanned to determine isotopes, and if necessary, these swabs will undergo neutron activation analysis to determine the fissile content.

3. EXPERIMENTAL RESULTS

The results presented below from the fuel rod nondestructive examination are for the fuel rod strain and visual examinations. Results are also presented from the crud measurements which include weight measurements and isotopic analysis of capsule swabs and capsule gas analyses. The results from the destructive examination of PH462-E3 are summarized.

3.1 VISUAL EXAMINATION

All eight rods were visually examined. The general surface condition of the H.B. Robinson fuel rods appeared to be unchanged. The fuel rods heated in air (both intact and defected) were similar in appearance to those heated in argon (Figures 4 and 5).

The general surface appearance of the four BWR fuel rods also appeared not to have changed. The artificial defect at the top of rod PH462-C5 did not appear to be fully open. When this hole was drilled into the cladding, indications were that the drill bit had penetrated through the wall. Debris may have been pulled into the hole upon removal of the drill bit. The defect located at the bottom showed significant enlargement and developed an axial crack emanating from both sides of the original hole (Figure 6). Also the top defect in rod PH462-D6 did not appear to be fully open to the atmosphere.
3.2 FUEL ROD STRAIN

Significant fuel rod strain was measured in the BWR PH462-C5 fuel rod. The crack size in PH462-C5 was asymmetrical extending 8.6 mm up the rod and 10.2 mm down the rod. Its width at the widest point was 1.3 mm. The original defect, which was 0.76 mm, increased to 0.25 mm neglecting the crack opening. The fuel rod expanded 17% at 90° from the crack and 7% at 30° almost parallel to the crack. There is no apparent fuel rod strain in the other three BWR fuel rods based upon the fuel rods' diameter measurements. However, although the fuel rod diameter indicated no strain, the middle defect in PH462-D6 slightly increased in diameter by 0.11 mm. The top defect was apparently closed, and measurements indicated nominal defect size.

The diameters for the PWR fuel rods did not change much from the nominal dimensions; they varied from 10.46 to 10.67 mm; slightly less than the nominal 10.72 mm. The smaller diameter may be attributed to cladding creep during steady-state irradiation in the reactor. These values are comparable to the values measured previously (Einziger and Cook, 1983; Einziger and Cook, 1984). Similarly, the defect sizes did not increase.

3.3 GAS RELEASE

Gas samples were taken from the six sealed fuel rod capsules (Table 4). The gas sample from PH462-E4 proved to be invalid because the valve on the capsule was found to be opened when the gas sample was taken. Air had apparently leaked into the argon-filled capsules, but this is consistent with a $2.5 \times 10^{-7}$ cm$^3$/s leak rate for 7,206 h. A higher than normal amount of CO$_2$ was found in HBR-G7 (0.39%) and in HBR-B8 (0.50=9%). The reason for this high CO$_2$ content is not known.
With PH462-D6, a significant amount of fission products was released in the capsule. These products were the long-lived isotopes of krypton and xenon. These gases may have been released from the grain boundaries as the fuel was oxidized. These products were not detected in the sealed capsules of the PWR fuel rods.

3.4 CRUD AND FUEL PARTICULATE RELEASE

Each of the stainless steel fuel rod capsules was swabbed with a cotton swab in order to capture any crud which may have fallen from the rod, or any fuel which may have been released from a breached fuel rod. Each of the individual filters and swabs was weighed to determine whether any fuel or crud was captured. The pre and posttest weight measurements are listed in Table 5. The negative values for the filters reflect balance drift for the size of loads being measured. The weight changes are insignificant. All of the swabs indicate a weight gain including the control swabs which were just placed in the cell during the time the capsules were being swabbed. The largest weight gain occurred with PH462-C5 that was breached. The weight gains of the other swabs do not appear to be significant compared to the control samples.

Various isotopes collected on the filters and swabs were determined by isotopic gamma scanning. These results are shown in Table 6. Only the 15 μm filter for the PWR fuel rod BO5-E7 exhibited cesium-137, but the other fission products attributed to fuel are not present. This measurement may be anomalous. The swab measurements indicate that some crud was found in all the capsules, and some fuel in capsules which contain defected fuel rods except BO5-G7. This rod was defected, but the isotopes indicate only crud. PH462-E5 is an intact rod, but the isotopes indicate the presence of fuel.
3.5 SUMMARY OF RESULTS OF DESTRUCTIVE EXAMINATION OF PH462-E3

The crack in PH462-E3 grew from 1.3 cm to 6.35 cm during the last 3,727 h at 502 K (Figure 7), and the cladding swelled to 16% from fuel oxidation. The largest strain occurred where the fuel oxidized to $\text{U}_3\text{O}_8$. Further away from the crack, less oxidation occurred. In sample TF at 7.0 cm from the artificial defect, the fuel was primarily oxidized to $\text{U}_4\text{O}_9$, but some $\text{U}_3\text{O}_8$ was present. The oxidation occurred intergranularly so that with even an unetched sample the grain structure was revealed by oxidation (Figure 8).

A particle which was released from the crack area was determined by X-ray diffraction to be $\text{U}_3\text{O}_8$. This particle was very easily pulverized into face powder, and contamination problems resulted just in handling this powder for X-ray diffraction measurements. Similar behavior was observed in metallographic preparation of the other samples containing oxidized fuel. Individual grains of fuel were pulled from the pellet during polishing and smeared across the cladding leaving a stain.

The fuel at the center defect was oxidized primarily to $\text{U}_4\text{O}_9$ with some $\text{U}_3\text{O}_8$. Correspondingly, the cladding deformation was small, on the order of 0.3%. The oxidized fuel in this area was also pulled from the fuel and stained the cladding.

4. DISCUSSION

In unlimited air, cracks were initiated from the artificial defects located at the ends of BWR fuel rods PH462-E3 and PH463-C5 rather than at the center. The defect at the top of PH462-C5 did not enlarge because the defect may have been blocked. A destructive examination will show the condition of this defect as well as the defects in other fuel rods.
The extent of the oxidation, which occurred in the BWR fuel rods, did not occur in the PWR fuel rods based on dimensional changes. Oxidation of PWR fuel will be determined during the ongoing destructive examination. This result may be in part due to a smaller plenum volume.

Although the BWR fuel rod PH462-D6 may not have oxidized as evidenced by the lack of fuel rod deformation, this defected fuel rod continued to release fission gases. Fission gas release from this rod was also measured during the second nondestructive examination, and may indicate open porosity in the fuel in PH462-D6 and some oxidation. Densification in the BWR fuel and the rod gap volume may be contributing factors to the extensive BWR fuel oxidation. Open intergranular porosity may offer channels for the transport of oxygen into the interior of a fuel pellet.

The crud release inside the capsules was very minimal, but the crud layer on both the PWR and BWR fuel rods is also very small, as evidenced by the visual examination of the fuel rods.

5. CONCLUSIONS

These tests have shown that at least with BWR fuel, storage of defected fuel rods in an unlimited air atmosphere at temperatures between 490 K and 502 K may result in cladding failure and loss of the fuel out of the breached cladding. The oxidation also appears to cause further fission gas release. Because of its age, the BWR fuel may be atypical of current fuel design and fabrication techniques. The oxidized fuel results in contamination of the surrounding area which will be difficult to decontaminate.

The storage of PWR fuel rods in unlimited air at 502 K is uncertain. A destructive examination is planned for the PWR rods as well as the BWR rods so that more information will be gained on the extent of the oxidation and the cause of fission gas release from PH462-D6.
Although both PWR and BWR fuel rods have a small amount of crud, it is not very heavy. However, some crud is removed from the fuel rods probably during insertion and removal from the capsules.

REFERENCES


Einziger, R.E. and Cook, J.A., August 17-18, 1983, First Interim Examination of Defected BWR and PWR Rods Tested in Unlimited Air at 229 C, presented at Workshop on Spent Fuel/Cladding Reaction During Dry Storage in Gaithersburg, Maryland.


**TABLE 1:**
**DEFECT LOCATIONS**

<table>
<thead>
<tr>
<th>Fuel Rod</th>
<th>Defect Number 1</th>
<th>Defect Number 2</th>
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<tbody>
<tr>
<td></td>
<td>Location</td>
<td>Orientation</td>
<td>Location</td>
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<tr>
<td>BO5-E7</td>
<td>14.5</td>
<td>90°</td>
<td>70.5</td>
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<td>BO5-G7</td>
<td>11.0</td>
<td>200°</td>
<td>81.0</td>
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<td>PH462-D6</td>
<td>20.5</td>
<td>90°</td>
<td>88.5</td>
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<tr>
<td>PH463-C5</td>
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<td>0°</td>
<td>93.0</td>
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</table>

a. Measured from the top of the fuel rod.
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<thead>
<tr>
<th>Reactor Type</th>
<th>Assembly and Rod Number</th>
<th>Capsule Atmosphere</th>
<th>Capsule Pressure (MPa)</th>
<th>Fuel Rod Condition</th>
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</thead>
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<tr>
<td>PWR</td>
<td>BO-5-G7</td>
<td>Ar/3% He</td>
<td>0.1</td>
<td>defected (2 holes)</td>
</tr>
<tr>
<td>PWR</td>
<td>BO-5-E7</td>
<td>Air</td>
<td>open capsule with filters</td>
<td>defected (2 holes)</td>
</tr>
<tr>
<td>PWR</td>
<td>BO-5-08</td>
<td>Ar/3% He</td>
<td>0.1</td>
<td>intact</td>
</tr>
<tr>
<td>PWR</td>
<td>BO-5-B8</td>
<td>Air/3% He</td>
<td>0.1</td>
<td>intact</td>
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<td>BWR</td>
<td>PH-462-D6</td>
<td>Ar/3% He</td>
<td>0.1</td>
<td>defected (2 holes)</td>
</tr>
<tr>
<td>BWR</td>
<td>PH-462-E4</td>
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<td>PH-462-E5</td>
<td>Ar/3% He</td>
<td>0.1</td>
<td>intact</td>
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<tr>
<td>BWR</td>
<td>PH-462-C5a</td>
<td>Air</td>
<td>open capsule with filters</td>
<td>defected (3 holes)</td>
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a. After 2nd furnace campaign, this rod replaced PH-462-E3.
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<td>Capsule Number</td>
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<td>B05-G7</td>
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<td>Component</td>
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<td>-----------------------------</td>
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<tr>
<td></td>
<td>60&lt;sup&gt;Co&lt;/sup&gt;</td>
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<tr>
<td>7K</td>
<td></td>
</tr>
<tr>
<td>PB-C5 Capsule</td>
<td>28.7 (0.09)&lt;sup&gt;a&lt;/sup&gt;</td>
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<td>HBR-B8 Capsule</td>
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<td>PB-D6 Capsule</td>
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<td>HBR-O8 Capsule</td>
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<td>HBR-G7 Capsule</td>
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<td>HBR-E5 Capsule</td>
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</tr>
<tr>
<td>-E5, -G7, -O8 Control</td>
<td>0.009 (0.002)</td>
</tr>
</tbody>
</table>

<sup>a</sup> Numbers in parentheses are 1-6 standard deviation.
FIGURE 1:
Dry Fuel Storage Furnace
FIGURE 2:
Thermocouple Train
Dry Fuel Storage Temperature History

Pigage 3

IEL 4 4215

Time (h)

Temperature (°C)

7-13-84 Final exam
9-13-83 2nd interim exam
7-8-83 1st interim exam
1-24-82 Fuel loaded
8-15-82 1-22-82
FIGURE 4:
Top Defect in Fuel Rod HBR-G7 Heated in Argon
FIGURE 5:
Top Defect in Fuel Rod HBR-E7 Heated in Air
FIGURE 6:
Crack in Fuel Rod PH462-C5, 158 in. from the Top
Cladding Defect after Two Furnace Campaigns

FIGURE 7:
Cladding Defect in PH462-E3 after Two Furnace Campaigns (S40772)
Oxidized Fuel
As Polished

FIGURE 8:
Oxidized Fuel As-Polished Condition (S40757)
AN OVERVIEW OF DRY INTERIM STORAGE RESEARCH AND DEVELOPMENT ACTIVITIES AT THE E-MAD FACILITY, NEVADA TEST SITE

J.B. Wright and H.H. Irby

Westinghouse Electric Corporation
Nevada Operations, U.S.A.

ABSTRACT

Research and development activities involving dry storage of spent nuclear fuel have been in process at the Nevada Test Site since 1978. The E-MAD facility, originally designed and built to support the nuclear rocket development program in the 1960s and early 1970s, has been utilized to demonstrate dry interim storage of spent fuel, and to perform tests, processing, and packaging which require special remote handling techniques. The dry storage demonstrations have included drywell, silo, and vault storage of spent nuclear fuel, as well as simulation tests using fuel assemblies and electric heaters. The facility is operated for the U.S. DOE Nevada Operations Office by the Waste Technology Services Division of the Westinghouse Electric Corporation.
1. **INTRODUCTION**

The Engine Maintenance, Assembly and Disassembly (E-MAD) facility, which is located at the United States Department of Energy's (DOE) Nevada Test Site, was built in the mid 1960's as a test facility for the then active space nuclear rocket program. In 1977, extensive modification was begun to convert the E-MAD to a large scale R&D facility for the advanced development and demonstration of the handling, packaging, and storage of spent commercial nuclear fuel. Facilities that were added for this particular purpose included a surface storage silo, (Figure 1), near surface storage drywells, (Figure 2), a fuel storage vault (Figure 3), and additional handling and packaging equipment that would be required to support these facilities.

During the modification of E-MAD it was also equipped with two very specialized pieces of equipment, the boiling water calorimeter (Figure 4) and the Fuel Temperature Test (FTT, Figure 5) apparatus. The calorimeter was designed to measure the heat output from a single fuel assembly by totally immersing the fuel in boiling water and measuring the change in boiloff rate due to the fuel.

The results of these data could then be used to verify the irradiation history of the fuel and also verify computer codes used to predict decay heat output of fuel assemblies. The FTT was designed to provide an elevated temperature, simulated storage environment for evaluation of fuel integrity performance under worst case storage conditions. The fuel could be subjected to high temperatures (up to 300°C) that could not be achieved with normal decay heat and storage configurations.

The E-MAD facility contains one of the largest hot cells in the world. A floor plan of E-MAD hot cell facilities is shown in Figure 6. The main Hot Bay measures approximately 140 feet long by 65 feet wide and 80 feet high with concrete walls and lead glass windows 5 to 6 feet
thick. All cask handling and fuel handling at E-MAD is performed in this cell. The cell is equipped with a 40 ton overhead crane and two 500 pound capacity wall mounted mechanical manipulator arms. The cell also has a complete complement of master slave manipulators as well as viewing devices such as thru-wall periscopes. The cell is accessible from the outside through a 20 foot wide by 40 foot high shielded door that allows passage of over-the-road tractor/trailer trucks or standard gage railroad rolling stock.

2. FUEL HANDLING AND STORAGE

During the past several years, spent fuel handling, packaging, and storage activities have been carried out in support of the Spent Fuel Handling and Packaging Program (SFHPP) and the Spent Fuel Test-Climax (SFT-C). During 1978 and 1979, 17 spent fuel assemblies with burnups between 20,000 and 30,000 MWD/MTM were received at E-MAD from Florida Power's Turkey Point reactor. Work performed on these fuel assemblies included: cask handling; receipt inspection of fuel assemblies; boiling water calorimetry of fuel assemblies; canisterization and closure welding; emplacement of fuel canisters in solo, drywell, or vault storage locations; elevated temperature testing; and data collection and documentation of storage conditions and histories.

During this period, E-MAD also provided all fuel handling and canisterization services for the SFT-C facility, a 1400 ft deep test mini-repository located in a granite formation on the Nevada Test Site. A summary of fuel handling activities at E-MAD, up to the current fiscal year, is shown in Figure 7. A detailed report on E-MAD fuel handling is contained in Reference 1.

In concert with fuel canisterization programs, E-MAD devoted considerable efforts to development of weld processes for canister closures. These included both the Tungsten Inert Gas (TIG) and Plasma
Arc Weld (PAW) processes. The only process actually used in remote operations for both the SFHPP and SFT-C programs was TIG.

3. CSFM SUPPORT

In Fiscal Year 1983, E-MAD began conducting additional fuel handling services in support of the Commercial Spent Fuel Management (CSFM) programs. This included cutting open fuel storage canisters that had been in storage for up to 4 years at E-MAD and SFT-C; characterizing the fuel through visual examination and photodocumentation using periscopes and video, surface contamination swipes, and calorimetry; removal of individual fuel pins for examination and storage for future detailed characterization; and retrieval of Material Interaction Test (MIT) capsules that had been emplaced in certain fuel assemblies.

In March of 1983, a long term elevated temperature FTT run using PWR fuel assembly B02 was initiated to simulate the storage of fuel in a metal cask, air atmosphere environment. The 20 zones of heat control in the FTT allow us to impose a temperature profile on the fuel from top to bottom that very closely matches that which one would expect under actual storage. Prior to insertion into the FTT, two fuel pins were removed from the B02 fuel assembly, examined, photographed, and then stored in inert gas for comparison with other pins from the assembly when the assembly is removed from the FTT at the termination of the test. The test began at a fuel centerline temperature of 275°C, with the temperature being reduced monthly to simulate fuel decay. The test is currently scheduled for an 18 month duration. Samples of the air atmosphere are taken monthly and analyzed for the presence of He and Kr-85 which would indicate a loss of cladding integrity during the test.

Other special fuel handling activities at E-MAD in 1984 involved the recovery of Material Interaction Test coupons that had been inserted in the control rod guide tubes of certain fuel assemblies at Battelle
Columbus Laboratories. These coupons were intended to document the radiation environment in the fuel assemblies during storage. The coupons were retrieved, identified, packaged and shipped to the Hanford Engineering Development Laboratories for analysis. Also, one of the storage canisters from the SFT-C was selected, sectioned, packaged and shipped to LLNL for analysis of radiation effects on the canister material during storage.

The E-MAD facility has contributed significantly to the success of the spent fuel programs in the U.S., its unique capabilities have been beneficially utilized by several DOE laboratories in the attainment of spent fuel program goals.

REFERENCE

Figure 1. E-MAD Sealed Storage Cask Silo
Figure 2. E-MAD Drywell Configuration
Figure 3. E-MAD Lag Storage Pit
Figure 4. E-MAD Boiling Water Calorimeter
Figure 5. Fuel Temperature Test Stand
Figure 7. Summary of Fuel Handling Activities at E-MAD
ABSTRACT

A program of tests was performed in the hot-cells at Chalk River Nuclear Laboratories to determine whether or not irradiated Zircaloy cladding rings containing starter cracks would crack further at temperatures relevant to dry storage when stressed and exposed to vapours of iodine or cesium/cadmium. The results showed that Zircaloy with low fast neutron fluences (<0.1 x 10^{24} n/m^2) would fail only at high stress intensities, above the defect thresholds previously established for unirradiated Zircaloy. Irradiated rings of Zircaloy with substantial fast neutron fluences (2.5 - 9.2 x 10^{24} n/m^2) were susceptible to failure when stressed to yield point with stress intensities <2.5 MPa\sqrt{m} in either iodine or cesium/cadmium. Furthermore, when exposed to either iodine or cesium/cadmium vapours, it was found that cracks had nucleated in about one-third of smooth irradiated specimens stressed to yield point - considerably in excess of the threshold stress (~140 MPa).

These results are highly pessimistic for dry storage conditions since:

(i) clad stress was unrepresentatively high in these tests compared with the stresses in spent fuel cladding which are typically a small fraction of the yield stress,
(ii) high concentrations of corrosants were used in these tests whereas in irradiated fuel rods iodine and cesium will be combined chemically with other species, and therefore not readily available to cause environmental cracking, and

(iii) other papers presented in this International Workshop have proved that irradiated fuel rods will not fail when stored for extended periods at a variety of temperatures.

This work was funded 50% by the Electric Power Research Institute (EPRI) and 50% by CANDEV. The latter is a joint funding arrangement between Ontario Hydro (OH) and Atomic Energy of Canada Limited.
1. INTRODUCTION

This work was performed to investigate whether or not iodine vapour or mixtures of cesium and cadmium vapours are capable of causing cracks to propagate in irradiated Zircaloy cladding at temperatures under consideration for spent fuel dry-storage. In particular we wanted to investigate the relationship between critical stress intensity for crack propagation and temperature and compare the findings with critical stress intensity-temperature curves previously established (1) in tests on unirradiated Zircaloy. We provided potentially harmful environments in each of the tests and studied the relative effects of either fixed amounts of the corrodants which were depleted during the tests or constant (higher) vapour pressures. In reviewing the results, it should be appreciated that (i) the corrodant concentrations were unrepresentatively high since the fission products iodine, cesium and cadmium may be combined sufficiently with other species in spent fuel rods that they are prevented from attacking the Zircaloy cladding, and (ii) clad stresses were also unrepresentatively high (310-720 MPa) compared with (a) anticipated clad stresses <100 MPa in dry storage and (b) the threshold stress of 140 MPa for self-nucleated environmental cracking.

2. EXPERIMENTAL TECHNIQUE

2.1 SPECIMEN PREPARATION

Rings, about 13 mm long, cut from CANDU fuel cladding of diameter 15.24 mm were split longitudinally on one side and stressed by opening the slot with a block of Zircaloy 19 mm wide. Thus the maximum tensile stress was generated at the inner surface of the tube opposite the inserted block. The specimens were individually identified by engraving a number close to the block in the region of lowest stress. Then the stressed rings were washed in methanol and air-dried.

Starter cracks were induced by immersing the stressed rings individually in a 1% iodine-methanol solution with an acoustic probe
coupled directly onto the block of each specimen to monitor crack initiation. The piezoelectric probe signal was filtered, amplified and fed into an oscilloscope. The number of acoustic emissions per second above a pre-set threshold was displayed on the cathode ray tube and the complete acoustic history was recorded in the oscilloscope memory.

When the acoustic emissions were sufficiently intense and frequent each stressed ring was removed from the methanol-iodine bath and immersed in water to terminate the cracking process. Following a 20 minute aqueous wash the stressed rings, now containing starter cracks, were rinsed in methanol and dried.

A thin blue interference film of oxide was next grown on the freshly created starter crack surfaces. This was achieved by anodizing individual specimens in 10% oxalic acid in distilled water for two to three seconds at a potential difference of 80 volts. The blue anodized oxide film enabled the depth of starter cracks to be determined when the rings were eventually broken apart during or following the tests. After further washing in methanol the rings were dried.

2.2 ENCAPSULATION OF STRESSED RINGS AND CORRODANTS

Two types of Pyrex capsules were designed and fabricated, one type having a side-limb inside the furnace hot zone and the other having a side limb to contain the corrodant outside the furnace. Vials with goose-neck break-seals and containing the required quantities of iodine, or cesium and cadmium, were placed in the side limbs. The Pyrex capsules were placed in the hot cell and the stressed Zircaloy ring specimens were introduced into the capsules, sixteen or seventeen per capsule. Then the capsules were clamped shut at an O-ring seal in the capsule body and a vacuum pump reduced the internal pressure to 0.1 Pa at which time the neck was flame-sealed to isolate the capsules and their contents. Release of the corrodant from the vial into the main capsule was performed by shaking such that the fragile goose-neck break seal on the vial fractured.
2.3 TESTING AT ELEVATED TEMPERATURES

The capsules containing mixtures of cesium and cadmium were placed entirely in the furnaces so that the vapour pressures of cesium and cadmium were controlled by the furnace temperature or the quantities of cesium and cadmium. The capsules containing iodine protruded through port holes in a specially designed heavily insulated furnace door so that the specimens inside the furnace reached the desired temperature while the side limb containing the iodine crystals was maintained at ambient temperature to control the iodine vapour pressure at 36 Pa inside the capsule. In tests A, B and C the side limbs contained an excess of iodine crystals so the entire tests were conducted at 36 Pa iodine pressure. Tests D, E and F began at 36 Pa iodine pressure but the side limbs contained only 10 mg of iodine and this was consumed during the test, reducing the iodine pressure to the equilibrium vapour pressure of iodine over zirconium iodide.

All the tests conducted with iodine were interrupted at 500, 1000, and 1500 h to inspect the specimens through the Pyrex capsule wall and the tests were terminated at 2000 h. The same procedure was employed for the Cs/Cd tests conducted at 373 K but at 473 and 573 K the corrodant attacked the seals causing premature termination of the tests. Table 1 shows the test matrix and a brief summary of results.

2.4 POST-TEST EXAMINATION

After testing, the capsules were opened and the rings that had remained intact throughout the test were reverse-bent, first by hooking the ends of the rings over two pins protruding from the jaws of a vice and opening the vice, then by bending into a strip with master-slave manipulators. This reverse-bending cracked open some of the unfailed specimens and the fracture faces were then examined together with the fracture faces of specimens that had failed during exposure to corrodant and stress. After measuring the starter crack depth indicated by blue anodized patches (starting from the edge that was
originally on the inside of the ring) the fracture faces were either sketched or photographed at a magnification of 12.5 X. All of the rings has been stressed to their yield point before the test and the stress relaxed throughout the tests. Thus pairs of stress values are given in Table 2, the stresses having been interpolated from previous measurements (1,2) according to fast fluence and temperature of testing. Two values of stress intensity $K_I = 1.1 \sigma \sqrt{\pi a}$ were then calculated using the stresses $\sigma$ at the start and finish of the test and the initial starter crack depth $a$ measured from the blue anodized patch on the fracture face.

3. MATERIALS TESTED

A total of 199 stressed rings were encapsulated and tested with either iodine or cesium/cadmium vapours. The rings had been cut from the cladding of fuel elements from twenty-two bundles irradiated in the NRU, NPD, Douglas Point and Pickering reactors and had received fast neutron fluences up to $9.2 \times 10^{24}$ n/m$^2$ ($E > 1$ MeV). Table 2 shows values of Zircaloy fast fluence, starter crack depths, stresses and stress intensities which were typical of all tests performed.

Hydrogen and deuterium analyses were performed on Zircaloy cladding specimens using the technique of isotope-dilution mass-spectrometry. The analyses are accurate to $\pm 6\%$ and the precision of the measurements was confirmed by analyzing equal weights of National Bureau of Standards zirconium samples containing $107 \pm 3$ and $19 \pm 2$ mg/kg of hydrogen. In the specimens irradiated in CANDU power reactors the total equivalent hydrogen concentration was taken as the hydrogen concentration plus half of the deuterium concentration (Table 3). The values 20-130 mg/kg are representative of our wealth of past experience with fuel cladding from bundles with similar histories. Other properties of the same cladding material were described elsewhere. (1)
4. EXPERIMENTAL RESULTS

4.1 TESTS CONDUCTED IN IODINE VAPOUR

In the tests conducted in Capsules A, B and C at an iodine vapour pressure of 36 Pa the results were not strongly dependent on temperature in the range 373-473K (Table 1). In each case, 10 or 11 of the 14 rings with starter cracks failed and one of the smooth specimens failed in each capsule. Figure 1 (lower part) shows the distribution of stress intensities in failed and intact specimens in Capsules A, B and C as a function of test temperature.

Similar results were obtained from the tests conducted in Capsules D, E and F in which the total contained iodine was restricted to 10 mg except that the defect rates were slightly lower. Both of the smooth specimens in Capsule D (373 K) failed but the smooth specimens in Capsules E and F remained intact (Table 1). One ring failed in the final testing period (1500-2000 h) in each of Capsules D and F showing that the environments remained aggressive despite the limited quantity of iodine, depleted by reaction with the Zircaloy. The distributions of stress intensities and temperatures in specimens from Capsules D, E and F are shown in Figure 1 (upper part). The stress intensities were about the same as in Capsules A, B and C and the results were also similar, showing that cracks were able to propagate at stress intensities <2.3 MPa \( \sqrt{m} \) under yield stress conditions.

4.2 TESTS CONDUCTED IN CESIUM/CADMIUM VAPOURS

Capsules G, H and I contained stressed Zircaloy ring specimens and excess quantities of cesium and cadmium contained within the hot zone of the furnace such that the vapour pressures were governed by the equilibria at the furnace temperatures viz:
<table>
<thead>
<tr>
<th>Capsule</th>
<th>Temperature (K)</th>
<th>Cs Vapour Pressure (Pa)</th>
<th>Cd Vapour Pressure (Pa)</th>
</tr>
</thead>
<tbody>
<tr>
<td>G</td>
<td>373</td>
<td>9 x 10^{-2}</td>
<td>66 x 10^{-6}</td>
</tr>
<tr>
<td>H</td>
<td>473</td>
<td>12</td>
<td>53 x 10^{-3}</td>
</tr>
<tr>
<td>I</td>
<td>573</td>
<td>263</td>
<td>6.6</td>
</tr>
</tbody>
</table>

A problem was encountered with the capsule seals at the higher temperatures and air leaks occurred at the higher temperatures in Capsule H (between 1000 and 1500 h) and Capsule I (between 0 and 500 h). Consequently, to avoid further oxidation of the fracture surfaces of the rings the testing of Capsules H and I was terminated prematurely. However, the failure rates were very high in these three tests (Table 1) and five out of seven of the smooth rings cracked. The results appeared not to be strongly dependent on temperature or vapour pressures of corrosive species in the ranges indicated above. Figure 2 (lower part) shows the distributions of stress intensities in failed and intact specimens in Capsules G, H and I. Low stress intensity defects were particularly prevalent at 373 K and the three smooth specimens tested at 573 K defected even though the test in Capsule I was terminated at 500 h.

Fewer specimens failed in Capsules J, K and L (Table 1) in which the total contained cesium and cadmium were restricted to 50 mg and 1.14 mg respectively. At the highest test temperature an air leak was experienced in Capsule L in the period 500 - 1000 h. The other two capsules did not leak but Capsule K was not returned to the furnace after 1500 h because the companion Capsule H (same temperature, more corrodants) had begun to leak and we did not want to risk obliterating the fracture surfaces of rings in Capsule K. Capsule J remained in the furnace at 373 K for the scheduled 2000 h period. Less than 50% of the specimens failed in these tests showing that the reduced availability of cesium and cadmium had a significant effect relative to failure rates in Capsules G, H and I. Only one of the smooth specimens cracked, indicating that crack nucleation was dependent on the supply of cesium and cadmium.
Figure 2 (upper part) shows distributions of stress intensities and temperatures for the tests in Capsules J, K and L. The evidence of low stress intensity failures appears less in these tests but the evidence is somewhat obscured by the fact that six of the seven failed specimens from Capsule L had been too badly oxidized to determine starter crack depths, hence the initial stress intensity was unknown.

4.3 SCANNING ELECTRON MICROSCOPY

Five specimens from Capsules A, B, C, G and K were examined in detail in the scanning electron microscope to identify fracture surface features. The evidence is summarized in Table 4.

The fracture surfaces of the specimens that had been cracked in iodine at 373–473 K consisted entirely of stress corrosion type features, elongated cusping being more common than intergranular, transgranular and fluting features (Table 4).

The fractures of the two specimens that had been exposed to cesium/cadmium vapours differed according to test conditions. The fracture of Specimen 23 tested at 373 K had nucleated by intergranular Cs/Cd metal vapour embrittlement to a depth of 25 μm. The rest of the fracture appeared to have been assisted by hydride cracking. However, the fracture of Specimen K13 (tested at 473 K) was dominated by Cs/Cd metal vapour embrittlement (Table 4). The equivalent hydrogen concentrations in Specimens 23 and K13 were 45 and 130 mg/kg respectively.

5. DISCUSSION

5.1 INFLUENCES OF FAST FLUENCE AND EQUIVALENT HYDROGEN CONCENTRATION ON BATCH FAILURE SUSCEPTIBILITIES

Table 1 showed that the incidence of failure of specimens in Capsules A through L was not strongly dependent on temperature or
corrodant. However, some batches of cladding appeared to be more susceptible than others to failure so we evaluated batch susceptibility against two properties of the cladding viz. fast neutron fluence and equivalent hydrogen concentration.

Figure 3 shows batch susceptibility plotted against fast fluence. No trend is evident except that for the lowest fast fluence (<0.1 \times 10^{24} \text{ n/m}^2) the batch susceptibility is very low, indeed we will see later (Section 5.2) that Batches 1150, PSA and 1110 contributed 22 specimens, 20 of which remained intact. The rest of the data of Figure 3 show that in the range 2.5 - 9.2 \times 10^{24} \text{ n/m}^2 there is no trend of increasing or decreasing susceptibility with fast fluence.

Figure 4 shows batch susceptibility plotted against equivalent hydrogen concentration. There is no strong trend. A possible weak dependence is suggested by the dashed lines but the locations of data points PSA, 1150 and 1110 were probably determined by their low fast neutron fluences rather than hydrogen content. The overall conclusion is that batch defect susceptibility was random with respect to equivalent hydrogen concentration, the value being 65 \pm 35%.

5.2 Threshold Stress Intensity

Figure 5 shows a histogram of the incidence of failure of rings as a function of fast fluence. The low incidence of failure (9%) at very low fast fluence (<0.1 \times 10^{24} \text{ n/m}^2) contrasts with the 66% failure rate at fast fluences in the range 2.5 - 9.2 \times 10^{24} \text{n/m}^2. This observation agrees with the results from earlier experiments (3) which showed a step-increase in the susceptibility to cracking of smooth stressed Zircaloy rings cut from irradiated cladding between 0.8 and 2.1 \times 10^{24} \text{ n/m}^2 (E>1 \text{ MeV}) in tests at 573 K in iodine vapour. Therefore, threshold stress intensity will be considered separately for the two fast fluence ranges.
5.2.1 $K_T$ Threshold for Cladding with Low Neutron Fluences

A graph of stress intensity versus test temperature (Figure 6) was plotted to compare the behaviour of the rings irradiated to low fast fluences with the threshold curves previously established in tests on unirradiated Zircaloy. Only two irradiated specimens out of twenty-two failed at fluences less than $0.1 \times 10^{24} \text{ n/m}^2$, both of them having been exposed to 10 mg iodine: Specimen J8 at 423 K and Specimen J4 at 473 K. These were both well above the defect thresholds established using unirradiated cladding (Figure 6). Three other data points representing intact irradiated specimens fell into the defect field which was consistent with the previous tests which showed populations of intact and defective unirradiated specimens in the defect field.

There were seventeen intact specimen data points that fell below the threshold curve but for clarity we have plotted only the highest two representing Specimens F4 and J1. These two specimens had the same stress intensity and temperature but different corrosive environments (Figure 6) placing them at the edge of, and just below, the defect thresholds. Low power optical fractography revealed that environmental cracking had advanced slightly beyond the tips of their starter cracks.

The crucial point from these tests is that the low fluence specimens in Capsules A through L behaved consistently with our previous tests (1) on unirradiated Zircaloy showing that the test environmental conditions were equivalent. Thus the low stress intensity defects encountered in specimens with higher fast fluences (next section) should be regarded seriously and cannot be ascribed to an uncontrolled test variable.

5.2.2 $K_T$ Threshold for Cladding with Fast Neutron Fluences

$2.5 \times 10^{24} - 9.2 \times 10^{24} \text{ n/m}^2$

Figures 1 and 2 showed that rings irradiated to $2.5 - 9.2 \times 10^{24} \text{ n/m}^2$ were able to fail at stress intensities less than 2.3 MPa $\sqrt{\text{m}}$. 
Indeed eleven out of thirty smooth specimens failed without starter cracks (Table 1) so cracks were able to self-nucleate at the yield stress for the following test conditions:

<table>
<thead>
<tr>
<th>Condition</th>
<th>Temperature</th>
</tr>
</thead>
<tbody>
<tr>
<td>36 Pa I$_2$</td>
<td>373K, 423K, 473K</td>
</tr>
<tr>
<td>$\leq$36 Pa (10 mg) I$_2$</td>
<td>373K</td>
</tr>
<tr>
<td>Excess Cs/Cd</td>
<td>373K, 573K, and 373K</td>
</tr>
<tr>
<td>Limited Cs/Cd</td>
<td>373K</td>
</tr>
</tbody>
</table>

In all other tests defects occurred at yield stress with stress intensities less than 2.5 MPa$\sqrt{m}$ with the exception of Capsules H and L which had suffered air leaks.

We mentioned in Section 4.3 that Specimen 23 had cracked in Cs/Cd at 373 K by metal vapour embrittlement to a depth of 25 $\mu$m. Further fracture had been assisted by hydride cracking. At the applied stress of 645 MPa the critical stress intensity at which hydride cracking was triggered was 6.3 MPa$\sqrt{m}$ in good agreement with the literature on this effect. In all other specimens examined in detail (Table 4) the environmental cracking had been sufficiently fast that hydride cracking had not occurred. Figure 7 shows a hydrogen solubility curve for Zircaloy which indicates that the combinations of equivalent hydrogen concentrations and test temperatures of the five specimens examined by scanning electron microscopy were such that hydrides would have been precipitated in all cases. Indeed Specimen 23 contained the least hydrogen (Figure 7) but this was the only one where hydride cracking had occurred on account of the low test temperature and correspondingly low Cs/Cd vapour pressures.

6. CONCLUSIONS

Irradiated Zircaloy rings with fast fluences in the range 2.5 - 9.2 x 10$^{24}$ n/m$^2$ ($E > 1$ MeV) failed readily (14% to 59% defect rates) at very low stress intensities (<2.5 MPa$\sqrt{m}$) when stressed to yield point in either iodine or cesium/cadmium vapours. The defect
rates at low concentrations of $I_2$ and Cs/Cd were 28% and 14% and at
high concentrations 44% and 59% respectively. Eleven of the thirty
smooth specimens defected showing that the environmentally-induced
cracks could nucleate as well as propagate when the Zircaloy was
stressed to the yield point.

Specimens of Zircaloy tubing with very low neutron fluences
and a variety of stress intensities behaved just as expected from
previous laboratory tests on unirradiated Zircaloy rings. Above the
defect thresholds there was a mixture of intact and failed specimens
and beneath the defect threshold, 17 out of 17 specimens remained
intact. (These tests are of crucial importance because they show that
low $K_I$ defects in irradiated rings with higher fluences must be
attributed to the effects of irradiation rather than uncontrolled
anomalies in the test environments.)

In tests performed in iodine vapour there was no strong effect
of test temperature in the range 373-473 K on the incidence of defects
but reducing the amount of iodine from 36 Pa (constantly replenished)
to 10 mg (initially 36 Pa but depleted throughout the test) decreased
the failure rate in 2000 h from 70% to 53%.

In tests performed in cesium/cadmium vapours there was a
doubling of the defect rate on increasing the test temperature from 473
to 573 K in the tests done at the equilibrium vapour pressures of the
Cs and Cd vapours. The incidence of defects was insensitive to
temperature in the range 373-473 K irrespective of the quantities of
Cs/Cd in the test capsules and also in the range 473-573 K but only
when quantities of Cs and Cd were limited. Reducing the availability
of Cs and Cd from the equilibrium vapour pressures to 50 mg Cs plus
1.14 mg Cd reduced the defect rate from 76% to 46%.

The cracks that propagated in Cs/Cd vapours resembled those
cracked in $I_2$ vapour when the fracture surfaces were studied after
testing.

The incidence of defects was not a strong function of the
equivalent hydrogen concentration in the specimens in the range 20-130
mg/kg. Hydrides were rarely observed on the fracture surfaces because
iodine - or cesium/cadmium-induced cracking was faster than hydride
cracking in most test conditions.
REFERENCES


<table>
<thead>
<tr>
<th>CAPSULE</th>
<th>TEMP.</th>
<th>STRESS</th>
<th>ENVIRONMENT</th>
<th>CRACKS</th>
<th>NUMBERS TESTED</th>
<th>INCIDENCE OF SPECIMEN FAILURES</th>
<th>CUMULATIVE TOTALS FAILED</th>
</tr>
</thead>
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<td>A</td>
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<td>14</td>
<td>9</td>
<td>2</td>
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<td>8</td>
<td>1</td>
</tr>
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<td>473</td>
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<td>14</td>
<td>8</td>
<td>2</td>
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<td>14</td>
<td>8</td>
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<td>H</td>
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<td>Yield</td>
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<td>2</td>
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<td>I</td>
<td>573</td>
<td>Yield</td>
<td>12 Pa Cs</td>
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<td>15</td>
<td>7</td>
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<td>Yield</td>
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<td></td>
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<td>6.6 Pa Cd</td>
<td>No</td>
<td>3</td>
<td>0</td>
<td>-</td>
</tr>
</tbody>
</table>

Footnotes:  
(1) See Table 2 for actual values  
(2) Test Terminated at 1500 h  
(3) Test Terminated at 500 h  
(4) Test Terminated at 1000 h  
* = Air Leaks
**TABLE 2:**
CAPSULE A TEST DETAILS: 36 Pa I₂, 373 K

<table>
<thead>
<tr>
<th>SPECIMEN IDENTITY</th>
<th>BUNDLE IDENTITY</th>
<th>FAST FLUENCE E &gt; 1 MeV (10²⁴ n/m²)</th>
<th>STARTER CRACK DEPTH (μm)</th>
<th>STRESS (MPa)</th>
<th>STRESS INTENSITY (MPa ( \sqrt{m} ))</th>
<th>TIME TO FAIL (h)</th>
</tr>
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<tr>
<td>J1 71</td>
<td>1110</td>
<td>&lt;0.1</td>
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<td>57</td>
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*SCC Crack Depth = 73 μm Beyond Starter Crack

NF Stands for Not Failed
TABLE 3:
RESULTS OF HYDROGEN AND DEUTERIUM ANALYSES ON ZIRCALOY BUNDLES

<table>
<thead>
<tr>
<th>BUNDLE IDENTITY</th>
<th>HYDROGEN CONCENTRATION mg/kg</th>
<th>DEUTERIUM CONCENTRATION mg/kg</th>
<th>TOTAL EQUIVALENT HYDROGEN CONCENTRATION mg/kg</th>
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<td>1110</td>
<td>24</td>
<td>64</td>
<td>56</td>
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</tr>
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</table>

(1) $H_2 + \frac{1}{2}D_2$ is the total equivalent hydrogen concentration
<table>
<thead>
<tr>
<th>CAPSULE ID</th>
<th>SPHENON ID</th>
<th>BUNDLE ID</th>
<th>CONCENTRATION (mg/l)</th>
<th>FAST NEUTRON FLUX (10^17 n/m²)</th>
<th>TEST ENVIRONMENT</th>
<th>TEST TEMPERATURE (K)</th>
<th>STARTER STRESS (MPa)</th>
<th>STRESS INTENSITY (MPa/m)</th>
<th>TIME TO FAILURE (HOURS)</th>
<th>INTER-GRANULAR</th>
<th>TRANS-GRANULAR</th>
<th>ELONGATED</th>
<th>FLUTING</th>
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TABLE 4:
SCANNING ELECTRON MICROSCOPY: DETAILS OF SPECIMENS, TESTS AND FRACTURE
FIGURE 1:
Stress intensities in the rings exposed to iodine vapour and whether or not the specimens failed (Capsules A through F).
FIGURE 2:
Stress intensities in the rings exposed to cesium and cadmium vapours and whether or not the specimens failed (Capsules G through L). The data are incomplete for Capsules H, I and L because some fracture features were obliterated by oxidation.
Batch defect probability in all tests - random environment, temperature and stress intensity. Beside data points we have indicated bundle identities from which the batches of rings were taken.
FIGURE 4: Overall batch suscepitibility to cracking in tests conducted in capsules A through L as a function of equivalent hydrogen concentration.

OVERALL BATCH SUSCEPTIBILITY %

EQUIVALENT HYDROGEN CONCENTRATION (mg/kg)

\[= H_2 + 0.5 D_2 \]
Figure 5:

Fast Fluence (E > 1 MeV) x 10^12 n/m^2

Numbers of Rings

Intact

Failed
Stress intensities relative to the threshold curves previously established\(^{(1)}\) for unirradiated specimens. All these specimens (tested in Capsules A through L) had very low fast neutron fluences (<0.1 x 10\(^{24}\) n/m\(^2\)) and behaved essentially as unirradiated material. Fifteen other specimens (not shown) had stress intensities below the thresholds and did not defect.
FIGURE 7:
Hydrogen solubility curve for Zircaloy showing that the combined hydrogen concentrations and test temperatures of the five specimens examined by SEM were such that hydrides were precipitated in all cases.
A PROBABILISTIC PREDICTION OF THE
ALLOWABLE DRY STORAGE CLADDING TEMPERATURE

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and
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ABSTRACT

A procedure was developed to correlate spent fuel dry storage temperature to probability of clad breaching in 100 years and under isothermal conditions. Probabilistic distributions of spent fuel parameters, namely, fission gas release, pressure and chemical environment, were assessed and used as inputs to models of potential failure mechanisms: creep-rupture, cladding oxidation and stress corrosion cracking. Since current practical dry storage experience does not directly demonstrate the required 100 year safe life, the results are intended to help in design and in the licensing process for light water reactor spent fuel dry storage facilities.

(1) Graduate Research Assistant; presently with Integrated Device Technology, Santa Clara, California
(2) Professor (Research)
1. INTRODUCTION

1.1 The Importance of Dry Storage

Light Water Reactor (LWR) spent fuel has been discharged to water pools to remain there in wet storage in order to dissipate its decay heat. Wet storage is a safe and demonstrated technology that has been operating effectively for 40 years (Johnson, Jr. et al, 1982, 1983; E.R. Johnson Assoc, 1981).

Projections (Figure 1) show that the number of discharged fuel assemblies will surpass the present wet storage capability, in the U.S., by 1985 (Johnson, Jr et al 1982, 1983). Reprocessing has no prospects to be implemented in the near future (E.R. Johnson Assoc 1981). Dry storage is regarded as an attractive option for interim storage, namely, for 100 years, for having the following advantages (Johnson Jr et al, 1982).

* Cooling can be completely passive, which precludes concerns regarding interruption of coolant flow.

* Cover gas control and monitoring requirements are less extensive for dry storage than for wet storage.

* Some dry storage concepts can be added in modules, precluding large initial capital cost.

* It is feasible that less secondary radioactive wastes could be generated as compared with wet storage facilities.
1.2 The Necessity of a Probabilistic Model for Allowable Spent Fuel Storage Temperature

Current practical experience comes from laboratory tests and dry storage demonstrations, as summarized in Tables 1 and 2, respectively. Laboratory tests spanned temperatures from 100 to 570°C, and test times from 30 to 1300 days. Dry storage demonstrations spanned temperatures from -4 to 430°C, and times up to 8 years for low temperatures (Johnson, Jr et al. 1982, 1983; Kaspar et al. 1982). Only one rod failure has been reported (Johnson, Jr et al., 1983); it was stored at 270°C in air, but no assessment of the failure mode has been made. Although results of the tests are favourable, they do not directly demonstrate the required 100 years safe life. It is also unlikely that these experiments included enough fuel rods to account for low probability, high fission gas release and/or large initial flaw size cases. However, these low probability events are of concern in design and licensing. Thus, even with the benefit of additional field data, analytical predictions are still an essential element of the design and the licensing process for spent fuel storage (Miller et al., 1983).

2. SUMMARY AND CONCLUSIONS

The objective of this work is to infer quantitatively the fraction or percentage of spent fuel rods expected to fail within 100 years of dry storage as a function of the temperature that the cladding undergoes during dry storage ($T_s$). This output shall be useful as a design criterion either for storage facilities which can tolerate failure of very small fractions of cladding (e.g., $P_f$ less than $10^{-3}$ or $10^{-4}$), or for facilities which can stand a reasonable
number of failures (e.g., \( P_f \) less than 0.1). Only Zircaloy cladding was considered in the present work because it represents the majority, more than 95 percent, of the fuel which has to be stored.

To achieve this objective, probabilistic distributions for cladding conditions, prior to dry storage, were assessed to best represent the U.S.'s spent fuel inventory. Parameters such as fission gas release, pressure and iodine concentration were considered.

For temperatures up to 400\(^\circ\)C, which is the maximum being considered for dry storage (Blackburn et al, 1978), there are three potentially limiting cladding failure modes: oxidation, creep rupture and stress corrosion cracking (SCC) (Miller et al, 1983).

Having the probabilistic distributions for cladding parameters as inputs to models for failure by SCC (Cubicciotti, et al 1978), creep rupture (Blackburn et al, 1978) and oxidation (Kendall, 1955; Gulbransen et al, 1958), we could get the following results:

*For storage in inert atmospheres, SCC is the predicted limiting mode for temperatures below 340\(^\circ\)C, and creep rupture is the limiting mode above such temperature. \( P_f \) is about 0.1 for storage at 370\(^\circ\)C and about 10\(^{-3}\) between 190 and 340\(^\circ\)C. Lower \( P_f \) such as 10\(^{-4}\) can be obtained by using 150\(^\circ\)C as the storage temperature.

*For storage in atmospheric air, the results are similar to the ones for inert atmospheres for temperatures up to 320\(^\circ\)C. Up to this temperature, SCC is the limiting mode, but for higher temperatures oxidation takes place and will cause cladding failure before 100 years (\( P_f \)=1).

Although different modes of failure were treated separately, for simplicity, ignoring possible synergistic effects, these results are conservative because of the assumptions that have been made for
evaluating spent fuel rod conditions, and because of the inherent conservatism of the applied models. Six rods tested at $510^\circ C$ at Battelle Northwest Laboratory did not breach after almost one year of storage, whereas, the model would predict failure for all six rods. Not much can be said about most of the current dry storage experience because either the conditions are too mild or the time in storage is too short in comparison to 100 years. Hence, we conclude that the predictions of this work are compatible but probably conservative with respect to the current dry storage experience.

3. ASSESSMENT OF CLADDING CONDITIONS BEFORE DRY STORAGE

For the sake of simplicity, only data of prepressurized spent fuel used in Pressurized Water Reactors (PWR) was used. This is expected to be a conservative approach since pressures involved in such claddings are of the order of a few tens of atmospheres, whereas for unpressurized spent fuel, pressures involved are of the order of a few atmospheres (EPRI, 1982), so that stresses involved in prepressurized spent fuel are expected to be one order of magnitude higher than in unpressurized ones.

3.1 Fission Gas Release (fgr)

A sample of 51 prepressurized PWR rods was obtained to statistically represent the U.S.'s spent fuel inventory. It is constituted by three rods from Point Beach, six from Calvert Cliffs, 15 from Oconee (EPRI, 1982) and five from Turkey Point (Davis et al, 1979). Also, 22 rods from Zorita Reactor were included because they were fabricated by a domestic producer, and also because they were high
enrichment rods with extended burnups with possible future application (Balfour, 1982). These data, presented in Table 3, were rank ordered and plotted as cumulative % of rods versus %fgr (Figure 2); one can see that all rods have %fgr ≤ 24%.

The probability of failure in 100 years (θ) is related to the %fgr and, consequently to the cumulative % of rods. Suppose that at some temperature T a rod with 24%fgr, which corresponds to % rods equal to 100 percent (Figure 2), fails after 100 years. Since we have 51 rods, and only one has 24%fgr, the probability of failure is 1/51 or approximately 2 percent. Using the same reasoning for other %fgr, we can see that:

\[ \theta = 98 - \%\text{rods (}%fgr\text{)} \], \hspace{1cm} \theta \text{ in } \%.

Therefore, as the value of %fgr increases, %rods increases (i.e., a larger fraction of rods will have %fgr lower than that value). On the other hand, the higher the %fgr needed to cause failure under some condition, the lower will be the probability of failure, θ, as we expect from the previous expression.

Corresponding to the Weibull approach to statistical strength of materials, the probability axis was converted to log (-ln(1-θ)). The %fgr axis was converted to log (-ln(1-%fgr/100)) in order to provide a compatible scale, namely one stretching from -∞ at 0%fgr to +∞ at 100%fgr (Miller, 1983), as shown in Figure 3. Detailed calculations for Figures 2 and 3 are shown in Appendix A-1. In Figure 3, an extrapolation to 50% fgr, with a resulting θ o. 10^{-2}, was done to represent transient conditions, as in "bumping" experiments. Such extrapolation should be conservative since data from fuel rods operated under normal reactor conditions indicated an upper bound of 24%fgr, even for high-enrichment, high burn-up Zorita rods.
3.2 **Hoop Stress** ($\sigma_h$)

To determine the $\sigma_h$, we will assume, for convenience that all cladding is typical 14 by 14 PWR type with a diameter ($2r$) of 11.4 mm and a wall thickness ($t$) of 0.635 mm. Since $2r > t$, we write:

$$\sigma_h = \frac{P \times r}{t} = 9.0 \times P$$  \hspace{1cm} (1)

To convert $P$ to fission gas release, we will adopt a similar procedure to one used previously (Miller et al., 1983). Data from Maine Yankee Reactor (Fuhrman et al., 1976) indicate that 13 percent fgr corresponds to 119 cm$^3$ of fission gas at STP. For a typical void volume of 35 cm$^3$, the extra pressure $P$ is, therefore, equal to 0.262 * %fgr ($P$ in atm) or 0.0262 * %fgr ($P$ in MPa). Since prepressurization level is about 3.8 MPa at 25°C,

$$P = (3.8 + 0.062 \times (%fgr)) \times T / 298$$  \hspace{1cm} (2)

or

$$\sigma_h = (0.0008 \times (%fgr) + 0.115) \times T$$  \hspace{1cm} (3)

By combining the fgr distribution in Figure 2 with equations 2 and 3, the distribution of pressure and hoop stress (Figure 4) was derived from the distribution of %fgr.

3.3 **Iodine Concentration** ($I_2$)

In the fuel rod environment, iodine is mainly combined as cesium iodide (CsI) because of the large availability of Cs (Cubicciotti, 1978). Radiation may decompose iodine compounds under in-reactor conditions (Cubicciotti, 1976), yielding fresh $I_2$ which is an active element in the SCC process of the cladding, but radiation levels in dry storage are orders of magnitude lower: less than $10^6$ R/h gamma versus
$10^9$ R/h gamma, in-reactor, plus a high neutron flux (Johnson, Jr et al, 1983). However, it has been shown that there is another mechanism of getting reactive iodine, which may possibly occur in a spent fuel environment: decomposition of CsI can take place at sufficiently fast rates in the presence of $\text{UO}_2$ and Mo/Mo$_2$, giving rise to free iodine and Cs$_2$Mo$_4$O$_{10}$ (Hofmann et al, 1983).

Despite the great diversity of views on the subject, concentration estimates must be made in order to make quantitative predictions of limiting storage temperatures. Conservative assumptions, such as having iodine released from CsI at the cladding ID to the same extent as estimated for the in-reactor case, had to be made. By doing this, the following expression was obtained (Miller et al, 1978):

$$I_2 \text{ (kg/m}^2\text{)} = 0.2 \times f_{bu} \times f_{fgr} \quad (4)$$

A typical burnup is of approximately 30 000 MWD/MTU, which corresponds to a fractional burnup ($f_{bu}$) of 0.03. For the sake of simplicity, $f_{fgr}$ is assumed to have the value of 0.03 for all fuel rods. This is indeed a good assumption since for high burnup rods ($f_{bu}$ in the order of 0.05), $f_{fgr}$ can be of the order of 0.2 (Table 3), giving rise to iodine concentrations in the vicinity of (or greater than) the saturation level for SCC ($10^{-3}$ kg/m$^2$), hence the results will not change, by using $f_{bu} = 0.03$ when it is indeed 0.05.

By combining the $f_{fgr}$ distribution in Figure 2 with equation 4, the iodine concentration distribution was obtained (Figure 5).
4. FAILURE BY CREEP RUPTURE

Limiting dry storage temperatures for unirradiated cold worked stress relieved (CWSR) Zircaloy 4 (used in PWR's) were calculated by Blackburn et al in 1978, with respect to creep-rupture. Zircaloy creep data correlated by the Larson-Miller parameter yielded curves of 2/3 minimum stress for rupture for expected lifetimes ranging from 1 to 100 years (Figure 6). A maximum allowable cladding temperature of 380°C was recommended for a typical condition PWR fuel rod, corresponding to an internal pressure of 3.8 MPa at 25°C.

There might be some concern about the fact that irradiated cladding has different mechanical properties compared with those of unirradiated Zircaloy, from which the data used had been derived. It is possible, however, that the time for rupture is not significantly altered by irradiation. Neglecting primary creep effects, we can write \( t = \epsilon/\dot{\epsilon} \). The rupture ductility (\( \epsilon \)) will be decreased due to irradiation effects, but the strain rate will also decrease due to irradiation strengthening, and the net effect might be only a small change in time to rupture (E.R. Johnson, Assoc, 1981, Blackburn et al, 1978).

There is evidence that the criterion is conservative for irradiated cladding. Six irradiated PWR rods did not fail after being tested by Battelle Northwest Laboratory at 510°C, with hoop stresses higher that 90 MPa, as estimated using equation 3 for 0% fgr (\( \phi = 100\% \)). This is above the 2/3 minimum stress for rupture for 1 year lifetime (which is less than 50 MPa as shown in Figure 6). Hence, the statistical analysis predicted that all six rods would have breached before 1 year of storage, but they were reported without failures after approximately 1 year of storage (Einziger et al, 1982).
Superimposing lines from expression 3 in Figure 6, we can obtain the maximum allowable temperature for failure by creep-rupture after 100 years for different % fgr (Figure 7). This plot used in conjunction with the probabilistic distribution of %fgr (Figure 3) yields the probabilistic distribution of allowable temperature for failure by creep-rupture after 100 years storage (Figure 8). Detailed calculations are presented in Appendix A-2. Again, the dotted line is included to represent possible transient effects.

5. FAILURE BY OXIDATION

The uniform oxidation process of Zircaloy follows classical laws. Initially a thin, tight and adherent oxide film forms and the weight gain increases non-linearly with time; oxidation rate decreases with time because the increasing oxide layer promotes an increasing oxidant diffusion path. The process extends to approximately 10 μm, and is called "parabolic" oxidation because Δw ∝ t^2, where Δw is the weight gain. After that, the oxide layer cracks and even spalls, due to stress fields caused by the difference in volume between the zirconium and its oxide, and the penetration of oxidants is facilitated. During the latter phase, which is called linear oxidation, the rate is controlled by the oxidation chemical reaction at the metal surface.

Our failure criterion is oxidation through 50 percent of the wall thickness; hence, linear oxidation is dominant throughout the process, and will be the only one to be considered. Using oxidation data from Gulbransen, 1958 and EPRI, 1982, radial oxidation rates have been determined as a function of temperature and extrapolated to temperatures lower than 300°C (Figure 9) (Miller et al, 1983). Assuming a constant typical wall thickness of 0.635 mm, the temperature for oxidation in 100 years, in atmospheric air, was determined: 325°C.
6. FAILURE BY STRESS CORROSION CRACKING

Iodine is generally agreed to be the cause of failure by stress corrosion cracking of fuel rods during reactor operation. This type of failure, called pellet-cladding interaction (PCI), is caused by the combined action of the chemical attack of the internal surface of the cladding, and of tensile stresses arising from fuel expansion during power ramps (Cubicciotti, 1978). In dry storage conditions, there will be iodine formed during the fission process and tensile stresses from both prepressurization and fission gases. Therefore, SCC must be regarded as a possible mode of failure of spent fuel during dry storage.

A simple and intuitive way of modeling SCC is by using linear elastic fracture mechanics (LEFM), since SCC involves the propagation of a sharp crack through the wall of the cladding. The crack growth rate is shown to be function of stress intensity at the crack tip ($K_I$). It was determined experimentally that the rate of crack growth was proportional to $K_I^4$ at high $K_I$ values (Cox et al, 1977; Videm et al, 1979), as shown in Figure 10. Unfortunately, this simple law holds only for initially cracked Zircaloy specimens with flaw size ($c_o$) larger than 0.1 of the wall thickness ($t$), or $c_o/t > 0.1$. One can easily notice that for $c_o = 0$, this approach predicts erroneously that no failure should occur; for $c_o < 0.1$, life predictions based on LEFM will be unrealistically large (Miller et al, 1973). Another drawback of this method is the lack of information of the initial flaw size distribution for dry storage, which is a necessary input.
6.1 Approach Using SCCIG

A physical-phenomenological model, SCCIG, which stands for Stress Corrosion Crack Initiation and Growth, was developed by Miller and Tasooji to account for the various relevant inputs and physical processes of SCC of Zircaloy (Tasooji, 1978). This model accounts for local element failure by intergranular SCC, transgranular SCC, ductile rupture, changes in the local stress and strain distribution due to crack formation and growth, and iodine penetration along the crack. Detailed description of the model is presented elsewhere (Miller et al., 1983; Tasooji, 1978). Inputs to the model include: nominal hoop stress \( \sigma_h \), cladding temperature \( T \), fast neutron fluence, initial flaw size \( c_0 \), clad thickness \( w \) and iodine concentration \( I_2 \). Time for failure is the resultant output.

The clad thickness will be the same 0.635 mm that we have used before. The fast neutron fluence will be assigned the value of \( 2.15 \times 10^{25} \) n/m\(^2\), which is the saturation limit, according to data presented by Bain et al., 1979. Irradiation increases the SCC susceptibility of Zircaloy up to the saturation limit, beyond which the susceptibility remains unchanged (Tasooji, 1982). Normal fast neutron fluences for irradiated fuel cladding are higher than the saturation limit, according to data from Franklin et al., 1983.

To define the initial flaw size, we will make use of a result from previous SCCIG calculations. It was shown that if a given batch of spent fuel had had a probability of fuel failure in each rod of less than 1 percent, then in 99 percent of the intact fuel rods any incipient crack would be smaller than 20 percent of the wall thickness; in the remaining 1 percent of the rods there would be incipient cracks greater than 20 percent of the wall (Miller et al., 1983). This means that for a given set of input variables, i.e., for a given probability of occurrence namely, for a given \( I_2, w, \) fluence and hoop stress, if one uses
c/o = 0.2 to calculate the allowable storage temperature, the overall probability of failure for a unknown c/o is the probability of occurrence divided by 100, since storage conditions and incipient crack size are independent events. Hence, for c/o = 0.2 and for a given $\varnothing$, we have all the variables to calculate time of failure as a function of temperature, via SCCIG. Using temperatures corresponding to failure in 100 years for different $\varnothing$'s, we can get the temperature distribution for SCC failure in 100 years. Results are shown Figure 11 and detailed steps taken to do this are presented in Appendices A-3 and A-4.

7. RESULTS AND DISCUSSIONS

7.1 Oxidation

As we commented previously, the allowable temperature for storage under infinitely available air is 350 C. This result is not probabilistic because it is independent of the assumed probabilistic variables. This result is also expected to be conservative, because there will be only a limited amount of oxygen when air is the cover gas in storage casks; this will prevent oxidation to go much further and cause cladding failure (Blackburn et al, 1978).

7.2 Creep-Rupture

Maximum allowable temperatures for 100 year isothermal dry storage with respect to creep-rupture are shown in Figure 8. It varies from 332 C at probability of failure per rod, $\varnothing$ equal to 0.1% (including possible transient effects) to 372 C at $\varnothing = 14\%$ (note the $\varnothing = -\ln(1/\varnothing)$, for low values of $\varnothing$).
These results are expected to be conservative for several reasons:

*Use of only prepressurized rods and inclusion of high enrichment rods to evaluate the pressure distribution.

*Conservatism from adopting 0 probability of occurrence, of 1 percent for 50%fgr when possible transient effects were accounted in Figure 3. These first two factors resulted in a pressure distribution ranging from 3.97 to 6.42 MPa (Appendix A-2), which is higher than the normal range.

*Conservatism from using Blackburn's 2/3 minimum stress for creep-rupture criterion, shown in Figure 6, which was derived by using lower bound data for creep-rupture.

The only uncertainty of this criterion is that the data comes from mechanical properties of non-irradiated Zircaloy. Even so, there are reasons to believe that time to failure will not be greatly affected by irradiation, as we discussed in the section "Failure by Creep-rupture". We have also shown the method is indeed conservative compared to results of actual hot cell tests.

7.3 Stress Corrosion Cracking (SCC)

Allowable temperatures for 100 year isothermal storage with respect to failure by SCC are shown in Figure 11. Note that this plot was constructed for 20 percent precracked cladding; therefore, as we discussed earlier in the section "Failure by SCC", we have to divide 0 by 100 to get the actual probability of failure (since the probability of having a precrack greater or equal to 20 percent of the wall thickness is equal to 1 percent, and precrack size and storage conditions are
independent events); the final plot for claddings with unknown precrack size is shown in Figure 12. At 395°C, \( \theta \) is equal to 0.14 percent (14 out 10,000 rods are expected to fail by SCC before 100 years at 395°C). As the probability of failure, \( \theta \), decreases \( (\theta \sim -\ln(1/\theta)) \), \( I_2 \) increases from 66*10\(^{-5}\) kg/m\(^2\) to 102*10\(^{-5}\) kg/m\(^2\), at \( \theta = 0.07 \% \) and \( T = 195^\circ C \). The brisk change in the allowable temperatures is due to the fact that SCC is very sensitive to variations of the \( I_2 \) concentration. Once the concentration reaches the saturation level, 100*10\(^{-5}\) kg/m\(^2\), SCC becomes insensitive to further increases in concentration (Miller et al, 1981), which accounts for the vertical portion of the plot at \( T = 195^\circ C \). Further decrease in \( \theta \), for transient effects, is associated with an increase in internal pressure due to fgr, which causes further decrease in allowable temperatures down to 123°C, at probability of failure of 10\(^{-5}\) (one rod out of 100,000). Results for failure by SCC are expected to be conservative for the following reasons:

*Inclusion of high enrichment rods to evaluate fuel rod parameters before storage.

*Adoption of probability of occurrence of 1 percent for 50%fgr, due to possible transient effects, this resulted in conservative evaluations of pressure and iodine concentration.

*Additional conservatism in evaluating iodine concentration, assuming that 10 percent of the generated iodine is reactive, as discussed in Section 3.3.

7.4 Failure by Three Separate Mechanisms

Figures 13 and 14 show the superposition of different failure mechanisms for isothermal storage under air and under inert gas:
*For storage in inert atmospheres, SCC is the predicted limiting mode for temperatures below 340°C and creep rupture is the limiting mode above such temperature. $P_f$ is about 0.1 for storage at 370°C and about $10^{-3}$ between 190 and 340°C. Lower $P_f$ such as $10^{-4}$ can be obtained by using 150°C as the storage temperature.

*For storage in air, the results are similar to the ones for inert atmospheres for temperatures up to 320°C. Up to this temperature, SCC is the limiting mode, but for higher temperatures oxidation takes place and will cause cladding failure before 100 years ($P_f = 1$).

Although different modes of failure were treated separately for simplicity, ignoring possible synergistic effects, these results are conservative because of the conservative assumptions that have been made for evaluating spent fuel rod conditions, and because of the inherent conservatism of the applied models. Additional conservatism of the model comes from the fact that isothermal conditions do not prevail in actual storage conditions; Figure 15 shows how temperatures dropped for storage demonstrations due to radioactive decay.

7.5 Comparison Between Predictions of this Model with Dry Storage Experience

Current practical experience comes from laboratory tests and dry storage demonstrations, as summarized in Tables 1 and 2, respectively. Laboratory tests spanned temperatures from 100 to 570°C, and test times from 30 to 1300 days. Dry storage demonstrations spanned temperatures from -4 to 430°C, and times up to 8 years for low temperatures (Johnson, Jr et al, 1982, 1983; Kaspar et al, 1982).
Only one rod failure has been reported (Johnson, Jr et al, 1983); it was stored at 270°C in air, for 2 or 3 years, but no assessment of the failure mode has been made. For this temperature, both data and model are in agreement in that this is a low probability case. According to the model, we might expect that failure had been caused by large precrack size and/or high fission gas release.

REFERENCES


Figure 1. Distribution of projected U.S. LWR spent fuel inventory in various storage modes (from reference 1).
Figure 2. Fission gas release distribution for prepressurized PWR fuel rods.
Figure 3. Fission gas release distribution for prepressurized RWR fuel rods, with the dotted line extrapolating to represent low probability transient effects.
Figure 4. Estimate of pressure and hoop stress distribution at 25°C for prepressurized PWR spent fuel rods.
Figure 5. Estimate of the iodine concentration distribution for prepressurized PWR spent fuel rods.
Figure 6. Maximum allowable temperature, with respect to failure by creep-rupture, for different hoop stresses and lifetimes (from reference 6).
Figure 7. Maximum allowable temperature, with respect to failure by creep-rupture, for 100 year storage at different fgr levels.
Figure 8. Allowable storage temperature distribution, with respect to failure by creep-rupture.
Figure 9. Estimated oxidation behavior of Zircaloy cladding in air (from reference 5).
Figure 10. Experimental data on crack growth rate versus stress intensity, $K_I$ (from reference 8).
Figure 11. Allowable storage temperature distribution, with respect to failure by SCC. Only for precracked claddings with $c_o/w = 0.2$. 
Figure 12. Allowable storage temperature distribution, with respect to failure by SCC. For claddings with unknown precrack size.
CLADDING FAILURE BY THREE SEPARATE MECHANISMS

Figure 13. Allowable storage temperature distribution including three separate failure processes.
Figure 14. Allowable storage temperature distribution for storage under inert cover gas.
Figure 15. Temperature ranges and times for storage demonstrations involving Zircaloy clad fuel assemblies. The hypothetical curves are based on 24 1 kW PWR assemblies in a metal cask (from reference 1).
Table 1. Laboratory / hot cell tests involving Zircaloy-clad fuel rods (from reference 1).

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<td>Canada AECL/OM</td>
<td>Zircaloy-4</td>
<td>Rod capsules</td>
<td>220 to 250</td>
<td>Air</td>
<td>7,000 to 8,000</td>
<td>- 13 30</td>
<td>- 3,4,6 in progress</td>
<td>In progress Cladding defects elongated</td>
<td>27</td>
</tr>
<tr>
<td>Canada AECL/OM</td>
<td>Zircaloy-4</td>
<td>Whole rods</td>
<td>150</td>
<td>Moist air</td>
<td>5,000</td>
<td>- 8 1000</td>
<td>To be analyzed in 1984</td>
<td>In progress</td>
<td>5</td>
</tr>
<tr>
<td>FRG KWU</td>
<td>Zircaloy-4</td>
<td>Rod capsules</td>
<td>100</td>
<td>Ne + 1/2</td>
<td>0</td>
<td>- 550</td>
<td>- - No failures</td>
<td>-</td>
<td>3,4,6</td>
</tr>
<tr>
<td>FRG KWU</td>
<td>Zircaloy-4</td>
<td>Rod capsules</td>
<td>300</td>
<td>Ne + 1/2</td>
<td>0</td>
<td>- -</td>
<td>- - No failures</td>
<td>-</td>
<td>3,4,6</td>
</tr>
<tr>
<td>FRG/Ity ISpra/NUKEM</td>
<td>Zircaloy-2</td>
<td>Whole rods</td>
<td>400</td>
<td>Helium</td>
<td>33,000</td>
<td>14 9 190</td>
<td>To be analyzed in 1984</td>
<td>3/82 No cladding failures</td>
<td>9,4,6</td>
</tr>
<tr>
<td>USA BNL/NEEL</td>
<td>Zircaloy-4</td>
<td>Whole rods</td>
<td>480(e)</td>
<td>Helium</td>
<td>28,000</td>
<td>3 190</td>
<td>1-5 1979 0 cladding failures</td>
<td>7</td>
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</tr>
<tr>
<td>USA BNL/NEEL</td>
<td>Zircaloy-4</td>
<td>Whole rods</td>
<td>510</td>
<td>Limited air, Helium</td>
<td>28,000</td>
<td>3 325</td>
<td>1-5 1 0 cladding failures</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>USA BNL/NEEL</td>
<td>Zircaloy-4</td>
<td>Whole rods</td>
<td>510(e)</td>
<td>Limited air, Helium</td>
<td>28,000</td>
<td>2 135</td>
<td>1-5 1 0 cladding failures</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>USA EPRI/TVA/HEPL/ANL</td>
<td>Zircaloy-4</td>
<td>Rod capsules</td>
<td>250 to 350</td>
<td>Air</td>
<td>28,000</td>
<td>- 48 248</td>
<td>In progress</td>
<td>In progress</td>
<td>1</td>
</tr>
<tr>
<td>USA NRC/NL/NN/EGG</td>
<td>Zircaloy-2</td>
<td>Whole rods</td>
<td>230</td>
<td>Unlimited, argon</td>
<td>17,000</td>
<td>4 248</td>
<td>1-3 B/3(f) 1 cladding defect elongated</td>
<td>24</td>
<td></td>
</tr>
<tr>
<td>USA NRC/NL/NN/EGG</td>
<td>Zircaloy-2</td>
<td>Whole rods</td>
<td>230</td>
<td>Unlimited, argon</td>
<td>31,000</td>
<td>4 248 1-3 B/3(f) 1 cladding defect elongated</td>
<td>24</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

(a) Fission gas analyses 4 - destructive analyses 7 - N_2O trapping
2 - nondestructive rod tests 5 - sipping 8 - fission product migration
3 - visual inspection 6 - weighing

(b) Number of cover gas samples analyzed.
(c) Unirradiated rods; however, under test conditions, radiation would have been fully annealed.
(d) Unpublished data supplied by J. Fleisch, DWR, Hannover, FRG.
(e) Isotermal for 190 days; then four rods were transferred to 570°C.
(f) Phase 2 is in progress.
Table 2. Dry storage demonstrations involving Zircaloy clad-fuel (from reference 1).

<table>
<thead>
<tr>
<th>Country</th>
<th>Site</th>
<th>Fuel Type</th>
<th>Burnup, HW/MU</th>
<th>Cladding</th>
<th>Storage Concept</th>
<th>Time in Act Storage</th>
<th>No. of Assemblies</th>
<th>Dry Storage Period</th>
<th>Cover Gas</th>
<th>Maximum Cladding Temperature, °C</th>
<th>Integrity Verifications</th>
<th>No. of Rods</th>
<th>References</th>
</tr>
</thead>
<tbody>
<tr>
<td>Canada</td>
<td>MURREX UNR</td>
<td>6,000</td>
<td>Zircaloy Sito</td>
<td>0.5 to 4 yr</td>
<td>1975 -</td>
<td>Helium</td>
<td>120</td>
<td>120</td>
<td>1984</td>
<td>2-4</td>
<td>-</td>
<td>-</td>
<td>1,2,5</td>
</tr>
<tr>
<td>Canada</td>
<td>MURREX PWR</td>
<td>6,000</td>
<td>Zircaloy Sito</td>
<td>0.5 to 4 yr</td>
<td>1976 -</td>
<td>Helium</td>
<td>120</td>
<td>120</td>
<td>1984</td>
<td>2-4</td>
<td>-</td>
<td>-</td>
<td>1,2,5</td>
</tr>
<tr>
<td>FRG</td>
<td>KWO PWR</td>
<td>33,000</td>
<td>Zircaloy-4</td>
<td>10 Mw</td>
<td>0.5 to 4 yr</td>
<td>Helium</td>
<td>120</td>
<td>120</td>
<td>1984</td>
<td>2-4</td>
<td>-</td>
<td>-</td>
<td>1,2,5</td>
</tr>
<tr>
<td>USA</td>
<td>EMAU/NTS(P)</td>
<td>17,000</td>
<td>Zircaloy-2</td>
<td>6 to 10 yr</td>
<td>3 NTU</td>
<td>May 1983 until final disposal</td>
<td>Helium</td>
<td>180</td>
<td>-</td>
<td>2</td>
<td>-</td>
<td>-</td>
<td>5,3,7,12</td>
</tr>
<tr>
<td>USA</td>
<td>EMAU/NTS(P)</td>
<td>20,000</td>
<td>Zircaloy-4</td>
<td>2 to 3 yr</td>
<td>4</td>
<td>1/78</td>
<td>Helium</td>
<td>180</td>
<td>120</td>
<td>1983</td>
<td>No rod failures</td>
<td>-</td>
<td>1,10</td>
</tr>
<tr>
<td>USA</td>
<td>EMAU/NTS(P)</td>
<td>20,000</td>
<td>Zircaloy-4</td>
<td>2 to 3 yr</td>
<td>1</td>
<td>11/78 to 2/82</td>
<td>Helium</td>
<td>150</td>
<td>100</td>
<td>204</td>
<td>1.5</td>
<td>-</td>
<td>7/83</td>
</tr>
<tr>
<td>USA</td>
<td>EMAU/NTS(P)</td>
<td>20,000</td>
<td>Zircaloy-4</td>
<td>2 to 3 yr</td>
<td>12</td>
<td>12/79 to 4/83</td>
<td>Helium</td>
<td>220</td>
<td>-</td>
<td>2488</td>
<td>1.5</td>
<td>-</td>
<td>7/83</td>
</tr>
<tr>
<td>USA</td>
<td>EMAU/NTS(P)</td>
<td>20,000</td>
<td>Zircaloy-4</td>
<td>2 to 3 yr</td>
<td>230</td>
<td>3/83 to 9/84</td>
<td>Helium</td>
<td>230</td>
<td>-150</td>
<td>2244</td>
<td>1.5</td>
<td>-</td>
<td>7/83</td>
</tr>
<tr>
<td>USA</td>
<td>EMAU/NTS(P)</td>
<td>20,000</td>
<td>Zircaloy-4</td>
<td>2 to 3 yr</td>
<td>230</td>
<td>3/83 to 9/84</td>
<td>Helium</td>
<td>230</td>
<td>-150</td>
<td>2244</td>
<td>1.5</td>
<td>-</td>
<td>7/83</td>
</tr>
</tbody>
</table>

(a) 1 - Fission gas analyses
2 - Non-destructive tests on rods
3 - Visual inspection
4 - Destructive analyses
5 - Slipping.
(b) Number of cover gas samples analyzed.
(c) Date of last cover gas analysis.
(d) Rods in one basket were examined after 4 years.
(e) Unpublished data supplied by J. Fleisch, DDR, Hanover, FRG.
(f) Precharacterization: Slipping, profilometry; eddy-current; and oxide thickness measurements on 10 rods.
(g) See Appendix C for details of Turkey Point fuel assembly histories.
(h) One of 18 assemblies analyzed at hot cell; 5 assemblies at NTS have precharacterized rods; only interim visual examinations and fission gas analyses have been conducted since the NTS demonstrations began.
(i) Three PWR assemblies have been tested in the fuel temperature test (FTT) module, one at a time.
(j) Unpublished data supplied by A. R. Hale, WSD, and DOE/RVO.
Table 3. Fission gas release data for PWR spent fuel rods.

1) Turkey Point (ref. 26): 5 rods with \%fgr \ 0.31;  
    average burnup of 25,665 MWd/MTU.

2) Zorita (ref. 12): 22 rods; burnup/%fgr pairs are  
    given: *  
    \begin{verbatim}
    57500/20.9  3800/16.9  55000/13.2  54400/22.6  
    34900/11.7  40800/7.3  40500/12.4  39100/1.3  
    38600/0.7  39100/1.1  38900/2.0  39500/0.9  
    38800/0.4  39100/1.7  37700/0.2  49300/2.5  
    41100/9.1  30400/5.9  29600/1.0  21600/0.2  
    31200/0.2
    \end{verbatim}

3) Beach Point (ref. 11): 3 rods irradiated for 2 cycles;  
    having %fgr of 0.90 , 0.55 and 0.25 . **

4) Calvert Cliffs (ref. 11): 6 rods (1 cycle of irradiation)  
    %fgr: 0.71 0.64 0.33 0.35 , 0.27 0.34 . **

5) Oconee (ref. 11): 15 rods (2 cycles of irradiation)**  
    %fgr: 0.56 0.54 0.99 0.41 0.77  
    0.60 0.92 0.43 0.29 0.61  
    0.58 0.40 0.82 0.88

* Burnups in MWd/MTU.

** 1 cycle ~ 16,000 MWd/MTU  
2 cycles ~ 25,000 MWd/MTU  
3 cycles ~ 35,000 MWd/MTU (reference 27)
Appendix A-1. Distribution of fgr in PWR spent fuel rods; derived from table 3 and to be used in figures 2 & 3.

<table>
<thead>
<tr>
<th>fgr up to</th>
<th># of rods</th>
<th>% of rods</th>
<th>$\phi$</th>
<th>$\ln(1-\phi)$</th>
<th>$\ln(1-f_{gr})$</th>
</tr>
</thead>
<tbody>
<tr>
<td>(%)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>2</td>
<td>41</td>
<td>80</td>
<td>.22</td>
<td>.248</td>
<td>.0202</td>
</tr>
<tr>
<td>4</td>
<td>42</td>
<td>82</td>
<td>.20</td>
<td>.223</td>
<td>.0408</td>
</tr>
<tr>
<td>6</td>
<td>43</td>
<td>84</td>
<td>.18</td>
<td>.198</td>
<td>.0619</td>
</tr>
<tr>
<td>8</td>
<td>44</td>
<td>86</td>
<td>.16</td>
<td>.174</td>
<td>.0834</td>
</tr>
<tr>
<td>10</td>
<td>45</td>
<td>88</td>
<td>.14</td>
<td>.151</td>
<td>.105</td>
</tr>
<tr>
<td>12</td>
<td>47</td>
<td>92</td>
<td>.10</td>
<td>.105</td>
<td>.128</td>
</tr>
<tr>
<td>14</td>
<td>48</td>
<td>94</td>
<td>.08</td>
<td>.083</td>
<td>.151</td>
</tr>
<tr>
<td>16</td>
<td>48</td>
<td>94</td>
<td>.08</td>
<td>.083</td>
<td>.174</td>
</tr>
<tr>
<td>18</td>
<td>49</td>
<td>96</td>
<td>.06</td>
<td>.062</td>
<td>.198</td>
</tr>
<tr>
<td>20</td>
<td>49</td>
<td>96</td>
<td>.06</td>
<td>.062</td>
<td>.224</td>
</tr>
<tr>
<td>22</td>
<td>50</td>
<td>98</td>
<td>.04</td>
<td>.041</td>
<td>.248</td>
</tr>
<tr>
<td>24</td>
<td>51</td>
<td>100</td>
<td>.02</td>
<td>.020</td>
<td>.274</td>
</tr>
</tbody>
</table>

* $\phi = 1 - \text{fraction of rods} + .02$ was used because it was assigned $\phi = .02$ for the highest fgr rod, i.e., this rod was assigned a probability of occurrence of ~ 2% among the set of 51 rods.
Appendix A-2. Distribution of allowable storage temperature, relevant to failure by creep-rupture; data for figure 8.

<table>
<thead>
<tr>
<th>φ</th>
<th>-ln(1-φ)</th>
<th>% rods</th>
<th>%gr</th>
<th>$\sigma_h$</th>
<th>P</th>
<th>T(°C)</th>
</tr>
</thead>
<tbody>
<tr>
<td>.02</td>
<td>.020</td>
<td>100</td>
<td>24</td>
<td>.133T</td>
<td>4.40</td>
<td>365</td>
</tr>
<tr>
<td>.03</td>
<td>.030</td>
<td>99</td>
<td>21</td>
<td>.130T</td>
<td>4.30</td>
<td>367</td>
</tr>
<tr>
<td>.07</td>
<td>.073</td>
<td>95</td>
<td>17</td>
<td>.127T</td>
<td>4.21</td>
<td>368</td>
</tr>
<tr>
<td>.085</td>
<td>.089</td>
<td>93</td>
<td>15</td>
<td>.125T</td>
<td>4.14</td>
<td>370</td>
</tr>
<tr>
<td>.10</td>
<td>.105</td>
<td>92</td>
<td>12</td>
<td>.122T</td>
<td>4.04</td>
<td>371</td>
</tr>
<tr>
<td>.12</td>
<td>.128</td>
<td>90</td>
<td>11</td>
<td>.122T</td>
<td>4.04</td>
<td>371</td>
</tr>
<tr>
<td>.13</td>
<td>.139</td>
<td>89</td>
<td>10</td>
<td>.121T</td>
<td>4.01</td>
<td>372</td>
</tr>
<tr>
<td>.14</td>
<td>.151</td>
<td>88</td>
<td>9</td>
<td>.120T</td>
<td>3.97</td>
<td>372</td>
</tr>
<tr>
<td>1.0</td>
<td></td>
<td>2</td>
<td>0</td>
<td>.112T</td>
<td>3.71</td>
<td>377</td>
</tr>
</tbody>
</table>

| * .001 | .001 | 95.5  | .194T | 6.42 | 332  |
| * .010 | .010 | 50    | .155T | 5.13 | 352  |
| * .020 | .020 | 100   | .143T | 4.73 | 360  |
| * .039 | .040 | 98    | .133T | 4.40 | 365  |
| * .042 | .043 | 98    | .129T | 4.27 | 367  |

* data corresponding to extrapolation to account for transient effects; see figure 3 for details.

** stress and pressure in MPa, temperature T in Kelvin.
## Appendix A-3

Distribution of allowable storage
temperature, relevant to failure by SCC; data for figure 8.

<table>
<thead>
<tr>
<th>$\phi$</th>
<th>$-\ln(1-\phi)$</th>
<th>$N_{\text{rods}}$</th>
<th>$f_{\text{gr}}$</th>
<th>$\sigma_h$</th>
<th>$I_2$</th>
<th>$T,(^\circ\text{C})$</th>
</tr>
</thead>
<tbody>
<tr>
<td>*****</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>0.02</td>
<td>0.020</td>
<td>100</td>
<td>24</td>
<td>0.133T</td>
<td>144</td>
<td>195</td>
</tr>
<tr>
<td>0.03</td>
<td>0.030</td>
<td>99</td>
<td>21</td>
<td>0.130T</td>
<td>126</td>
<td>195</td>
</tr>
<tr>
<td>0.07</td>
<td>0.073</td>
<td>95</td>
<td>17</td>
<td>0.127T</td>
<td>102</td>
<td>195</td>
</tr>
<tr>
<td>0.085</td>
<td>0.089</td>
<td>93</td>
<td>15</td>
<td>0.125T</td>
<td>90</td>
<td>245</td>
</tr>
<tr>
<td>0.10</td>
<td>0.105</td>
<td>92</td>
<td>12</td>
<td>0.122T</td>
<td>77</td>
<td>275</td>
</tr>
<tr>
<td>0.12</td>
<td>0.128</td>
<td>90</td>
<td>11</td>
<td>0.122T</td>
<td>66</td>
<td>315</td>
</tr>
<tr>
<td>0.13</td>
<td>0.139</td>
<td>89</td>
<td>10</td>
<td>0.121T</td>
<td>58</td>
<td>345</td>
</tr>
<tr>
<td>0.14</td>
<td>0.151</td>
<td>88</td>
<td>9</td>
<td>0.120T</td>
<td>50</td>
<td>395</td>
</tr>
<tr>
<td>1.0</td>
<td></td>
<td>2</td>
<td>0</td>
<td>0.112T</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

| *0.001 | 0.001 | 95.5 | 0.194T | 573 | 123 |
| *0.010 | 0.010 | 50   | 0.155T | 300 | 150 |
| *0.020 | 0.020 | 100  | 36    | 0.143T | 219 | 170 |
| *0.039 | 0.040 | 98   | 24    | 0.133T | 144 | 195 |
| *0.042 | 0.043 | 98   | 20    | 0.129T | 120 | 195 |

* data corresponding to extrapolation to account for transient effects; see figure 3 for details.

**$I_2$ in $10^{-5}\text{kg/m}^2$.

*** for $c_o/w = 0.2$. For unknown precrack size, divide by 100.
Appendix A-4. SCCIG simulations to determine the temperature for failure in 100 years, for different conditions; data for appendix A-3.

Neutron fluence: $2.15 \times 10^{25} \text{ n/m}^2$

Pre-crack size, $c_0/w = 0.2$; wall thickness, $w = 0.635 \text{ mm}$

<table>
<thead>
<tr>
<th>$-\ln(1-\phi)/I_2 (\text{kg/m}^2)/\sigma_h (\text{MPa})$</th>
<th>$T (\text{°C})$</th>
<th>$t (\text{year})$</th>
<th>$T (\text{fail})$</th>
</tr>
</thead>
<tbody>
<tr>
<td>$0.020/144\times10^{-5}/.133T$</td>
<td>150</td>
<td>b.t.**</td>
<td></td>
</tr>
<tr>
<td></td>
<td>160</td>
<td>b.t.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>180</td>
<td>b.t.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>190</td>
<td>b.t.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>200</td>
<td>0.88</td>
<td>195</td>
</tr>
<tr>
<td>$0.030/126\times10^{-5}/130T$</td>
<td>200</td>
<td>0.88</td>
<td>195</td>
</tr>
<tr>
<td>$0.073/102\times10^{-5}/127T$</td>
<td>200</td>
<td>0.88</td>
<td>195</td>
</tr>
<tr>
<td>$0.089/90\times10^{-5}/125T$</td>
<td>220</td>
<td>b.t.</td>
<td></td>
</tr>
<tr>
<td>$0.105/77\times10^{-5}/122T$</td>
<td>240</td>
<td>b.t.</td>
<td></td>
</tr>
<tr>
<td>$0.128/66\times10^{-5}/122T$</td>
<td>200</td>
<td>b.t.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>220</td>
<td>b.t.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>240</td>
<td>b.t.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>280</td>
<td>b.t.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>300</td>
<td>b.t.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>310</td>
<td>b.t.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>320</td>
<td>0.003</td>
<td>315</td>
</tr>
</tbody>
</table>

** b.t. = below threshold stress for failure by SCC **

(over)
<table>
<thead>
<tr>
<th>$\frac{-\ln(1-\phi)}{I_2} \text{ (kg/m}^2\text{)}/\sigma_h$ (MPa)</th>
<th>$T(°C)$</th>
<th>$t(\text{year})$</th>
<th>$T(\text{fail})$</th>
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<tr>
<td>$139/58*10^{-5}/.121T$</td>
<td>340</td>
<td>b.t.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>350</td>
<td>0.001</td>
<td></td>
</tr>
<tr>
<td></td>
<td>360</td>
<td>0.001</td>
<td></td>
</tr>
<tr>
<td>$0.151/50*10^{-5}/.120T$</td>
<td>340</td>
<td>b.t.</td>
<td></td>
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<tr>
<td></td>
<td>360</td>
<td>b.t.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>380</td>
<td>b.t.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>390</td>
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<tr>
<td></td>
<td>400</td>
<td>0.0001</td>
<td>395</td>
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<tr>
<td>$0.020/219*10^{-5}/.143T$</td>
<td>165</td>
<td>b.t.</td>
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<td></td>
<td>175</td>
<td>0.88</td>
<td>170</td>
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<td>$0.010/300*10^{-5}/.155T$</td>
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<td></td>
<td>125</td>
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<td>123</td>
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SESSION 6

DRY STORAGE – FUEL BEHAVIOUR (II)

Chairman: R.D. Page
Atomic Energy of Canada Limited
URANIUM DIOXIDE FUEL OXIDATION IN AIR BELOW 350°C

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CEGB, Berkeley Nuclear Laboratories,
Berkeley, Gloucestershire, GL13 9PB,
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ABSTRACT

Irradiated CAGR (Civil Advanced Gas-cooled Reactor) fuel pins have been deliberately defected and heated in air at 240°C for times up to 27 days.

Fuel pins containing large defects (≥ 2 mm diameter hole) underwent oxidation forming U₃O₈ as evidenced by production and release of loose particulate after 6 - 18 days, and by clad swelling and defect extension after further oxidation to a total of 18 - 24 days. Defect extension occurred when the clad swelling exceeded 2.3%. Greater clad swelling has been observed when occurring at some distance from a defect.

Fuel pins containing small defects (< 0.1 mm) have shown no outward evidence of oxidation after heating for 24 days at 240°C.

Destructive examination of the fuel pins will be performed to assess the amount and distribution of oxidised fuel.
1. INTRODUCTION

A programme of work on the oxidation of UO$_2$ in air has been underway for a few years within the UK Central Electricity Generating Board. The basic features of the oxidation mechanism in air have been investigated using small samples of irradiated and unirradiated UO$_2$ with the high density fuel used in Advanced Gas Cooled Reactor fuel pins, and the results reported previously (Simpson and Wood, 1982, 1983).

The application of the information to pin behaviour is not straightforward in that the consequences of oxidation which are of concern, namely clad rupture and swelling, and release of fine particulate, are expected to depend on defect and pin geometry. A series of air oxidation tests has therefore been carried out on deliberately defected full length irradiated CAGR pins and form the subject of this paper.

Studies of the behaviour of deliberately defected CANDU (Boase and Vandergraaf, 1977) and LWR (Einziger, 1983) fuel are of limited relevance to AGR fuel. The AGR fuel pin (Boocock et al., 1982) is a ribbed stainless steel can, approximately 1 m long and 15 mm diameter, containing a stack of annular UO$_2$ fuel pellets which terminates in sintox insulator pellets (dimensions are given in Table 1). The fuel pin therefore differs from LWR and CANDU fuel in cladding material and in the presence of a centre void. Pressurisation of the cladding onto the fuel locks the can onto grooves in, typically, every fifth pellet, thus reducing the fuel-clad gap and preventing ratchetting of pellets. The grooves thus produced in the can are termed anti-stacking grooves (a.s.g.). 36 fuel pins are arranged in three rings within graphite sleeves to form a fuel element, 8 of which are linked together with a tie-bar to form the core section of a fuel stringer. In reactor creepdown of the cladding onto the fuel results in rapid loss of the fuel-clad gap with the higher temperature elements.
2. MATERIALS AND EXPERIMENTAL METHODS

2.1 FUEL PINS

The pins came from a stringer from Hinkley Point B AGR, discharged in June 1983, after 1580 days at power, with a mean burn-up of 15,561 MWD/te(U). Outer ring pins were selected from element 2 (pin burn-up 18,320 MWD/te) and element 6 (18,640 MWD/te) to provide samples of similar burn-up with and without a fuel-clad gap.

The condition of the pins at the start of the experiment were determined by normal post irradiation examination (PIE) of companion pins from the same elements. This included pin measurement (diameter and length), gamma scanning, radiography and gas release measurements.

2.2 INSERTION OF PIN DEFECTS

Pins were defected by drilling (type a) or mechanical piercing (types b, c, d). Type a defects were inserted using a 1/16" (1.6 mm) drill through the lower end cap in an off centre position. Type b and c defects were introduced by the piercing equipment used to obtain PIE gas release measurements. Type b was a small defect ~0.1 mm diameter; piercing was stopped as soon as fission gas was detected producing essentially a depression and cracking of the cladding. Type c was a large defect caused by complete penetration of the piercing tool, resulting in a hole of ~2 mm diameter. Both type b and c defects were located ~40 mm from the pin bottom, at the pellet 1- pellet 2 interface. The type d defect was a large hole, like type c, but situated near the pin centre, at the pellet-pellet interface adjacent to the 6th anti stacking groove.

Following defect insertion all fuel pins were placed in a storage container which was evacuated and back-filled with argon.
2.3 FURNACES

The furnaces were not designed with these \( \text{UO}_2 \) oxidation tests in mind, and had to work at the end of their temperature range with the result that the temperature gradient along the furnace tube was larger than desirable with a range of approximately 50°C. Attempts were made to calibrate the furnaces using steel rods, of similar dimensions to fuel pins, with temperature sensitive labels attached. All experimental temperatures quoted in this paper are estimates, based upon these calibrations, of the temperature at the defect locations. Further calibration and refinement of experimental temperatures will be attempted.

The furnaces each held three fuel pins vertically, spaced \( \sim 2 \) cm apart, each pin being \( \sim 4 \) cm horizontally distant from three chromel-alumel thermocouples. The vertical positions of these thermocouples corresponded to distances from the pin bottoms of 1.5, 44.7 and 85.2 cm for furnace A, and 6.9, 48.4 and 84.0 cm for furnace B. Temperature control was exercised using the thermocouple nearest to the pin defects. Each furnace was supplied with air (containing 200 - 1000 vpm \( \text{H}_2\text{O} \)), at a flow rate of \( 150 \text{ cm}^3 \text{ min}^{-1} \), which was exhausted through a 0.45 \( \mu \text{m} \) silver membrane filter.

2.4 EXPERIMENTAL METHODS

Before oxidation each fuel pin was subjected to overall visual examination and photography in the vicinity of the defect. The fuel pins were then exposed, three in each furnace, for successive periods of 6 days, completion of the experiment usually being decided by observation of swelling in one or more of the fuel pins in a furnace. After each 6 day oxidation period, and at completion, the fuel pins were unloaded from the furnace for visual examination and photography, to look for any evidence of clad swelling or defect extension. After completion of an experiment the furnace outlet filter was removed for subsequent analysis (initially by gamma spectroscopy) and the fuel pins stored in argon before final examination.
The final examination planned is pin mensuration or radiography followed by pin cutting. Fuel pins which appear, visually, unswollen will pass through a pin mensuration rig; this equipment can measure pin diameters with an accuracy of ± 12 μm and so will detect any clad strain exceeding ~ 0.08%. Visibly swollen fuel pins will be radiographed before pin cutting is decided upon, and the programme of pin cutting (both transverse and longitudinal sectioning) will be supported by sectioning of visibly unswollen fuel pins. Sections will be examined for evidence of oxidation in the bore, the fuel-clad gap, interpellet gaps, and intrapellet cracks.

3. RESULTS

The results of four completed pin oxidations are summarised in Table 2. The first experiment with furnace A (pins 2.23, 27, 21) was initially inspected after 3 days, with subsequent inspections at 6 day intervals; other experiments were inspected at regular 6 day intervals. In some cases (fuel pins containing large, type c or d, defects) during handling of the fuel pins at the visual examination and photography station powder was seen to fall from the defect; the time at which this occurred is recorded in the fifth column and ranges from 6 - 18 days. This powder production did, in some cases, precede any obvious clad swelling.

So far swelling has only been observed (after 18 - 24 days at approximately 240°C) in large defect pins, but oxidation of two small defect pins (2.32, 2.26) will be continued. The swelling behaviour observed with large defect pins has varied according to the temperature distribution. Fuel pins with type c defects have shown large swellings (5, 10%) between the first and second a.s.g., figure 1, at some distance (5 - 12 cm) from the defect where temperatures may have been lower. Fuel pins with mid-point (type d) defects oxidised in furnace A exhibited less swelling (2 - 8%) but this was at a shorter time and in two cases was associated with defect extension, figures 2 and 3.
In all cases swelling has been accompanied by bending or the fuel pin with the defect, generally, on the outside of the bend.

Results of the detailed pin mensuration, pin radiography and sectioning are not yet available.

4. DISCUSSION

$\text{UO}_2$ is not thermodynamically stable (with respect to $\text{U}_3\text{O}_8$) in air at any temperature below $\sim 1500^\circ\text{C}$, but its oxidation can be very slow. At the temperatures considered in this paper the initial oxidation product is $\text{U}_3\text{O}_7$; in unirradiated $\text{UO}_2$, and perhaps also in irradiated $\text{UO}_2$, oxidation to $\text{U}_3\text{O}_8$ is preceded by an incubation period. Because of a density decrease conversion of $\text{UO}_2$ to $\text{U}_3\text{O}_8$ is accompanied by:

i) Spalling of particulate.

ii) Volume expansion which may result in cladding strain, and ultimately defect extension enabling particulate release, in defected fuel pins.

The rate and extent of $\text{U}_3\text{O}_8$ formation depends not only upon the rate of reaction of $\text{UO}_2$ with air, but also upon the rate of oxygen transport through clad defects. Unirradiated $\text{UO}_2$, oxidised in air at $250^\circ\text{C}$, has been observed (Simpson and Wood, 1983) to produce $\text{U}_3\text{O}_8$ as loose powder after 8 - 14 days. Unclad fragments of irradiated $\text{UO}_2$ (12,400 MWd/te CAGR fuel) produced loose powder after 4 - 7 days at $250^\circ\text{C}$ and this enhanced reactivity of irradiated fuel has been observed at temperatures of 225 and 200°C (Bennett, Price and Wood, 1985). An incubation effect is also evident in work of Boase and Vandergraaf (1977) and Ainscough and Oldfield (1964) through weight gain measurements made during the oxidation of unirradiated defected fuel pins with small circular defects. Weight gain measurements have not been possible during this work, so we were restricted to recording the time before any powder production or swelling was evident and the amount of swelling occurring before defect extension. We particularly wanted to
observe whether the free space of the pellet bore could accommodate some of the fuel swelling, thereby reducing strain of the cladding. Also whether oxygen is transported through the bore and fuel-clad gap, and how the formation of $\text{U}_3\text{O}_8$ is distributed.

4.1 PARTICULATE PRODUCTION

Production and loss of loose particulate through clad defects has only been observed with the very large type c and d defects, after 6 - 18 days. Although the first examination of pin 2.29 did reveal loose powder after only 6 days oxidation considerable tapping of the fuel pin was required to dislodge the few particles observed, loss of particulate being more noticeable after a further 6 days oxidation.

The observation of loose powder production from irradiated CAGR fuel after 4 - 7 days at 250°C, coupled with the activation energy (121 kJ mol$^{-1}$) appropriate to the induction period, $t_p$, for powder production in unirradiated $\text{UO}_2$ (Simpson and Wood, 1983) would lead us to predict loose powder production from unclad fuel after 7 - 12 days at 240°C. This timescale coincides with that quoted above for loss of particulate from defected pins indicating that these large defects ($\sim 2$ mm holes) render the cladding ineffective in preventing oxidation and enable the fuel in the vicinity of the defect to behave as if unclad.

With the smaller type b defect however the cladding is evidently still providing considerable protection from oxidised fuel particulate production or loss, with none being evident at up to 24 days at 240°C. The results of the destructive examination are needed to characterise the small defects and determine whether or not oxygen ingress and fuel oxidation has occurred. Similarly it is not yet known whether oxygen transport past the sintox end-pellet has occurred in pellets with end cap defects.

Comparison of the times at which powder was observed between pins 2.29 and 6.21, and between pins 2.24 and 6.22 or 6.24 suggests that particulate loss from element 2 pins precedes that from element 6 pins. The results of the destructive examination of these pins should
determine whether this observation is connected with the presence of
the fuel-clad gap which still exists in element 2 pins but which has
disappeared by creep-down of the clad at the higher irradiation tempera-
tures experienced by element 6 pins.

4.2 DEFECT EXTENSION AND CLAD SWELLING

Defect extension and clad swelling have only been observed
with the large type c and d defects.

Fuel pins with mid-point defects heated in furnace A all
exhibited slight swelling (2 - 8%), and in two cases defect extension,
figures 2 and 3, after 18 days at ~ 240°C. These values of clad swell-
ing associated with defect extension may be compared with the 2% dia-
metral increase below which Novak and Hastings (1983) reported no
significant sheath splitting during the oxidation of defected CANDU
fuel elements, and with a value of 1% which Ainscough (1968) considered
to be the maximum local volume change which can be accepted by AGR
cladding without further damage.

Larger clad strains can be accommodated without clad rupture
if the swelling is located away from the original defect. This is
exhibited by fuel pins heated in furnace B which have exhibited (Figure
1) swellings of 5 and 10%, after 18 and 24 days oxidation at ~ 240°C,
at a distance of 5 - 12 cm from their type c defects. This demonstrates
easy transport of oxygen along a fuel pin, presumably through the bore
or intrapellet cracks since the effect has been observed with both
element 2 and element 6 fuel pins. Destructive examination of these
fuel pins will reveal where and how much oxidation has occurred. The
occurrence of swelling at a distance from the defect may be due to
undetected temperature gradients in the furnaces used in these experi-
mental studies. Differences of only a few degrees can be significant
through the incubation periods encountered before U₃O₈ production and
fuel swelling occurs. A similar remote swelling (5% at 10 cm distant
from a type c defect) was observed in pin 2.21 heated in furnace A
where large (3 K cm⁻¹) temperature gradients exist towards the pin ends.
Ainscough and Oldfield (1964) observed, with unirradiated deliberately defected (1 and 1.5 mm hole) WAGR fuel pins, that gross failure of cladding began after 21 - 25 days at 250°C equivalent to (again using the activation energy of 121 kJ mol\(^{-1}\)) 36 - 43 days at 240°C. Thus in this work with irradiated fuel the observation of swelling and defect extension after 18 - 24 days at 240°C indicates an enhancement due to irradiation similar to that observed with unclad samples in the time to powder production at 200 - 250°C (Bennett et al., 1985).

The aim of this work has been to determine time limits before the deterioration of fuel pin cladding, and to investigate oxidation within a pin to test theoretical models of oxidation occurring by internal oxygen diffusion. Boase and Vandergraaf (1977) and Novak and Hastings (1983) have both reported that at 250°C oxidation of an irradiated CANDU fuel element proceeds with an oxide front velocity of \(\sim 10^{-4}\) cm min\(^{-1}\) or \(\sim 1\) cm in 6 days. Such a process and rate is insufficient to explain the 7 cm length of swelling which occurred at 5 - 12 cm from the defect in pins 6.21 and 2.29.

5. FURTHER WORK

This phase of whole pin tests will conclude with an investigation of the effect of changes in oxidising atmosphere.

Three fuel pins containing type a, b and c defects will be oxidised in furnace B in an atmosphere of 5% O\(_2\)/95% N\(_2\). This is to check the effect of reduced oxygen pressures on the incubation times before powder production and clad swelling. It is relevant to a comprehensive programme using unirradiated material investigating the effect of oxygen partial pressure on the kinetics of UO\(_2\) oxidation.

Two fuel pins containing type d defects will be oxidised in 0.5% NO\(_2\)/99.5% air in furnace A. The radiolysis product NO\(_2\) has been reported (White et al., 1983) to enhance the rate of UO\(_2\) oxidation and promote UO\(_3\) production.
Interpretation of the results of this defect pin oxidation programme will be supported by oxidation of unclad samples (in the range 175 - 400°C) obtained from an element 6 pin from the same stringer.

6. SUMMARY OF OBSERVATIONS

1) Loose particulate has been observed to be released from CAGR fuel pins containing large defects (2 mm) after heating in air at 240°C for 6 - 18 days.
2) Further heating of CAGR fuel pins containing large defects has resulted in defect extension after 18 days at 240°C. Defect extension was promoted by clad swelling exceeding 2.3%.
3) Gross swelling (5 - 10%) after 18 - 24 days at 240°C has been observed at some distance (5 - 12 cm) from large defects.
4) When breached by smaller defects (< ~ 0.1 mm) the cladding offers more protection to the fuel with no powder being evident after heating for 24 days at 240°C.

ACKNOWLEDGEMENT

This paper is published by permission of the Central Electricity Generating Board. I would like to thank Dr. K.A. Simpson for her continued interest in this work.
REFERENCES


TABLE 1

FUEL PIN DATA (PRE-IRRADIATION)

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<th>Property</th>
<th>Value</th>
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<tr>
<td>Fuel density</td>
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<tr>
<td>Pellet diameter</td>
<td>14.48 (mm)</td>
</tr>
<tr>
<td>Centre bore diameter</td>
<td>5.08 (mm)</td>
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<tr>
<td>Fuel-clad gap</td>
<td>0.03 (mm)</td>
</tr>
<tr>
<td>Cladding thickness</td>
<td>0.38 (mm)</td>
</tr>
<tr>
<td>Pin diameter</td>
<td>15.24 (mm)</td>
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<tr>
<td>Pin length</td>
<td>977 (mm)</td>
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**TABLE 2**

RESULTS OF AIR OXIDATION OF DEFECTED CAGR FUEL PINS

<table>
<thead>
<tr>
<th>Fuel pin number</th>
<th>Furnace</th>
<th>Defect type</th>
<th>Defect temp. (°C)</th>
<th>Powder observed (day)</th>
<th>Time to swelling (s) or defect extension (e) (day)</th>
<th>Fuel pin condition</th>
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<tr>
<td>2.23</td>
<td>A</td>
<td>a</td>
<td>215 - 225</td>
<td>-</td>
<td>&gt; 27</td>
<td>Unswollen</td>
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<tr>
<td>2.27</td>
<td>A</td>
<td>b</td>
<td>215 - 225</td>
<td>-</td>
<td>&gt; 27</td>
<td>Unswollen</td>
</tr>
<tr>
<td>2.21</td>
<td>A</td>
<td>c</td>
<td>215 - 225</td>
<td>15</td>
<td>27 (s)</td>
<td>Swelling (5.3%) away from defect (5 - 12 cm)</td>
</tr>
<tr>
<td>6.23</td>
<td>B</td>
<td>a</td>
<td>240</td>
<td>-</td>
<td>&gt; 24</td>
<td>Unswollen</td>
</tr>
<tr>
<td>6.27</td>
<td>B</td>
<td>b</td>
<td>240</td>
<td>-</td>
<td>&gt; 24</td>
<td>Unswollen</td>
</tr>
<tr>
<td>6.21</td>
<td>B</td>
<td>c</td>
<td>240</td>
<td>12</td>
<td>24 (s)</td>
<td>Swelling (10%) away from defect (5 - 12 cm)</td>
</tr>
<tr>
<td>2.29</td>
<td>B</td>
<td>c</td>
<td>240</td>
<td>6</td>
<td>18 (s)</td>
<td>Swelling (4.8%) away from defect (5 - 12 cm)</td>
</tr>
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<td>B</td>
<td>a</td>
<td>240</td>
<td>-</td>
<td>&gt; 24</td>
<td>Unswollen</td>
</tr>
<tr>
<td>2.26</td>
<td>B</td>
<td>b</td>
<td>240</td>
<td>-</td>
<td>&gt; 24</td>
<td>Unswollen</td>
</tr>
<tr>
<td>6.24</td>
<td>A</td>
<td>d</td>
<td>240</td>
<td>18</td>
<td>18 (e)</td>
<td>Swelling (7.8%) and defect extension</td>
</tr>
<tr>
<td>6.22</td>
<td>A</td>
<td>d</td>
<td>240</td>
<td>18</td>
<td>18 (e)</td>
<td>Swelling (3.0%) and defect extension</td>
</tr>
<tr>
<td>2.24</td>
<td>A</td>
<td>d</td>
<td>240</td>
<td>12</td>
<td>18 (s)</td>
<td>Swelling (2.3%)</td>
</tr>
</tbody>
</table>

Defect types:

- a = end cap defect (1.6 mm)
- b = small defect near pin end
- c = large defect near pin end (~ 2 mm)
- d = large defect near pin centre (~ 2 mm)
FIG. 1. Pin 6-21, after 24 days at 240°C, showing 10% peak swelling between first and second a.s.g.
FIG. 2. Pin 6.22, after 18 days at 240°C, showing onset of defect extension associated with swelling of 3.0%.

FIG. 3. Pin 6.24, after 18 days at 240°C, showing defect extension associated with peak swelling of 7.8%.
OXIDATION OF UO₂ AT 150°C TO 350°C

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ABSTRACT

Tests were performed on non-irradiated UO₂ pellets from 150 to 350°C in atmospheric air and controlled environments and on spent light-water reactor (LWR) fuel fragments at 200°C and 230°C in atmospheric air to determine the variables that affect oxidation behavior under dry storage conditions. Additional tests are planned to develop design data on limiting temperature-time conditions for dry storage of spent fuel without monitoring for cover gas inertness or for abnormal storage events.

The weight of spent fuel fragments increased 50 to 100 times faster than the weight of non-irradiated UO₂ pellets at 230°C. Non-irradiated pellet fragments gained weight 5 to 7 times faster than non-irradiated pellets. The fragments simulated fuel fragmented by thermal gradients during reactor power changes. Low-density powder (U₃O₈) formed at 0.05% and 0.3% weight gain for non-irradiated pellets and fragments, respectively, but had not formed at 3% weight gain for spent fuel fragments with a burnup of 29,000 MWD/MTU. Canadian investigators had found that powder formed at intermediate levels of weight gain in CANDU spent fuel fragments with an approximate burnup of 8000 MWD/MTU.
The combined effects of the high rate of weight gain in spent fuel and the burnup dependence of weight gain at powder formation resulted in a minimum in a plot of the time for the onset of powder formation (the powder induction time) versus burnup. The minimum in powder induction time occurs at or below burnup levels typical of CANDU spent fuel and spent fuel at the ends of some LWR rods. The results are described in terms of thermal and neutron irradiation-induced changes in UO₂ pellet structure and chemical composition.

Other tests were performed at up to 275°C with spent fuel fragments and non-irradiated UO₂ pellets in moist nitrogen to determine the suitability of nitrogen as a cover gas. No measurable weight gain or visible physical changes occurred during the first 2 months of testing.
FOREWORD

This work was conducted by Pacific Northwest Laboratory (PNL), which is operated by Battelle Memorial Institute for the U.S. Department of Energy (DOE) under Contract DE-AC06-76RL01830. The program is under the direction of the Commercial Spent Fuel Management (CSFM) Program Office at DOE's Richland Operations Office. Funding and program management was provided by the Commercial Spent Fuel Management Program Office at PNL.

1. INTRODUCTION

Regulatory and design criteria (Code of Federal Regulations 1984) for dry storage of spent LWR fuel restrict radioactive releases to the environment. Spent fuel that is free of dispersible fuel particulates can be retrieved after extended dry storage with low risk of releases to the environment. The cladding effectively contains radioactive material through all phases of spent fuel management. Therefore, oxidation of spent fuel to low-density uranium oxide powders, that could enlarge cladding breaches and become dispersed during dry storage, should be prevented.

The potential for oxidation of $\text{UO}_2$ to $\text{U}_3\text{O}_8$ to propagate cracks in LWR fuel cladding has been known for nearly a quarter of a century. An oxidation process was developed for manufacturing $\text{U}_3\text{O}_8$ powders for fuel processing and use in decanning irradiated fuel rods as an early in spent fuel reprocessing (Bodine et al. 1961). The density of $\text{U}_3\text{O}_8$ is sufficiently lower than the density of $\text{UO}_2$ to cause long cracks to form in Zircaloy or stainless steel cladding. The split cladding permits $\text{U}_3\text{O}_8$ release.
To prevent the possibility of $U_3O_8$ or $UO_2$ powder formation, the temperature-time exposure of spent fuel in air can be maintained below the cumulative exposure required for cladding degradation by fuel oxidation (Gilbert et al. 1983; White et al. 1983; Wheeler 1983, 1984). Alternatively, if inert cover gases are used, they must not contain potentially damaging quantities of oxidizing constituents (Johnson and Gilbert 1983a, 1983b).

At the time the DOE/PNL studies were initiated, published results on phase diagrams and thermodynamics clearly indicated that low-density orthorhombic $U_3O_8$ and $UO_2$ were thermodynamically preferred to cubic $UO_2$, $U_4O_9$, and high-density tetragonal $U_3O_7$ under dry storage conditions. Therefore, $U_3O_8$ was expected to form, given sufficient time. However, oxidation studies were conducted for only short periods of time to identify optimum conditions for rapid powder formation for fuel processing. Therefore, the kinetic parameters were not adequately developed for extrapolation to extended storage periods under characteristic spent fuel storage conditions.

Extended U.S. tests and demonstrations are being sponsored by DOE, the U.S. Nuclear Regulatory Commission (NRC), the Electric Power Research Institute (EPRI), and the Tennessee Valley Authority (TVA) to determine conditions under which spent fuel can be stored without powder formation. Investigations of $UO_2$ oxidation behavior are also in progress in Canada and the United Kingdom. A summary of these programs is presented in Appendix A.

The DOE/PNL Commercial Spent Fuel Management (CSFM) Program is investigating the oxidation behavior of $UO_2$ to determine allowable temperature-time exposures for candidate spent fuel for interim dry storage design conditions under declining temperatures (White et al. 1983; Gilbert et al. 1983). Non-irradiated $UO_2$ tests were conducted to screen the significant variables because a similar test matrix with spent fuel would have been prohibitively expensive. The storage system environments during interim dry storage include variable moisture levels.
and a high gamma radiation field. Spent fuel covers a range of burnup levels, and some spent fuel will contain gadolinia. Consequently, tests with spent fuel (some containing gadolinia) were conducted to develop design criteria for storage of spent fuel. The design parameters must include both environmental and fuel conditions.

Nitrogen is a prospective acceptable cover gas for extended dry storage of spent fuel. Not only is it plentiful, but it also leaks slower through small orifices than helium, neon, or argon due to its large bimolecular diameter (CRC Handbook 1980; Perry and Chilton 1973). Its heat transfer properties are superior to those of argon or neon. Its thermal conductivity is below that of He, but its high density enables greater heat removal by convection compared with He. Concerns regarding the potential reaction between oxidants and the spent UO₂ fuel at the site of small cladding breaches are being addressed. Oxidants are produced by radiolysis of nitrogen and residual moisture remaining from cask drying operations and releases from water-logged fuel rods during dry storage (Johnson et al. 1983; Olander et al. 1982). Experiments were conducted with non-irradiated UO₂ pellets in moist nitrogen in a cesium-137 gamma field to determine the suitability of nitrogen as a dry storage atmosphere.

The remainder of this paper describes the experimental procedures and discusses the results of tests on non-irradiated pellets in oxidizing and moist nitrogen cover gases and irradiated fuel performed at PNL and other laboratories.

2. SUMMARY

Oxidation studies on non-irradiated UO₂ revealed a complex behavior that was dependent on atmospheric air aging time, surface area/volume ratio, cover gas moisture, UO₂ density, thermal cycling, temperature changes, radiolysis products in the cover gas (gamma
radiation), and gadolinia content of the UC$_2$. Irradiated UO$_2$ (spent fuel) displayed an additional dependence on burnup. The time to form low-density U$_3$O$_8$ was shorter for fuel with burnup levels between 0 and 15 MWd/MTU than for fuel with burnup levels above 15 MWd/MTU. The oxidation behavior of spent fuel was postulated to result from synergistic contributions from: 1) increased surface area (pellet fragmentation); 2) lattice defects induced by atom displacements, fission damage, transmutations, and fission products that enhanced the reaction of oxygen at UO$_2$ surfaces and grain boundaries; and 3) oxygen diffusivity. The combination of increased surface area from fragmentation and irradiation-enhanced surface reaction accelerated the rate of weight gain by factors up to 100 at 230°C. Lattice defects led to enhanced oxygen diffusion along short-circuited diffusion paths (grain boundaries, dislocations, subgrain boundaries, and open porosity).

Burnup-enhanced oxygen diffusion increased the level of weight gain in irradiated UO$_2$ prior to U$_3$O$_8$ formation. The time to reach the O/U ratio of 2.67 for U$_3$O$_8$ induction was delayed by the high diffusivity due to the low radial oxygen concentration gradient at the surface where U$_3$O$_8$ first forms in low diffusivity non-irradiated and low burnup UO$_2$. Consequently, the allowable temperature-time exposure of spent fuel to oxidizing conditions is dependent on fuel burnup, composition, and cooling time. Choices are consequently available for spent fuel management to optimize dry storage design characteristics such as temperature gradients to match spent fuel characteristics.

If the 1984 examination of Controlled Environment Experiment (CEX) Canadian spent fuel with breached cladding shows no significant degradation from fuel oxidation after 3 to 4 years of storage in air at 150°C (Ohta 1980), then temperatures up to 150°C should be acceptable for dry storage in air or for storage of spent fuel without monitoring of cover gases for inertness. Spent fuel entering storage at 150°C will remain at that temperature for only a short time due to declining decay heat. Furthermore, the CEX fuel is in a burnup range that appears to be more susceptible to U$_3$O$_8$ formation than higher burnup LWR fuels.
Tests to qualify nitrogen as a cover gas for extended dry storage of spent fuel have shown satisfactory compatibility between moist nitrogen and both non-irradiated pellets and spent fuel fragments. No measurable weight gain or visible physical changes occurred during the first 2 months at temperatures up to 275°C.

3. EXPERIMENTAL PROCEDURES

The experimental procedures for DOE/PNL tests performed on non-irradiated pellets and bare spent fuel fragments are summarized in this section. The procedures are described in greater detail by White et al. (1983). Oxidation tests conducted on non-irradiated UO₂ provided complementary information to spent fuel tests and minimized the test matrices required to develop design data for dry storage of spent LWR fuel.

3.1 NON-IRRADIATED PELLETS

Tests on non-irradiated pellets involved placing bare pellets in an air cover gas. The composition of the air was varied: atmospheric air, air with controlled moisture content, and air with 1 vol% NO₂ gas to synthesize an oxidizing radiolysis product. The pellets were periodically weighed and examined to determine the kinetics of oxidation and to detect physical changes, such as the formation of low-density powders (U₃O₈ and UO₃). Selected specimens were subjected to x-ray examination. Continuous weight measurements in a recording balance were used to monitor the rapid rate of weight gain above 250°C. The aging time of the pellets in ambient atmosphere following hydrogen sintering was investigated to resolve differences in oxidation behavior between pellets subjected to testing immediately after a hydrogen anneal and pellets stored in ambient atmospheric air prior to being tested.
Non-irradiated pellets were irradiated during oxidation testing in a 200,000 R/h cobalt-60 gamma facility to evaluate the effects of radiolysis of the cover gas on the oxidation behavior. Cover gases consisted of air with different moisture levels and 20°C dew point nitrogen.

3.2 IRRADIATED PELLET FRAGMENTS

Oxidation tests were performed with Point Beach spent fuel fragments (29,292 MWD/MTU burnup). Thermal stresses fragmented the UO₂ pellets during reactor operation. Most fragment edge lengths were between 3 and 5 mm. A small quantity of finer particles was mechanically generated during cladding removal. The size distribution of Point Beach fuel fragments in typical test specimens was reported by White et al. (1983). Fragment size distributions for H.B. Robinson, Point Beach, and Shipping port specimens are compared in Table 1. Open-ended 50-mm long segments of Point Beach fuel rods were included in the tests to evaluate the effect of decanning bare fragments.

Point Beach spent fuel fragments were heated to within 1°C in atmospheric air for recorded time intervals in convection ovens (White et al. 1983). One oven was located next to cesium-137 sources to generate a gamma field of 150,000 R/h. The gamma field in the other oven was reduced to approximately 600 R/h by shielding.

Spent fuel specimens were periodically weighed and visually inspected through 4-ft thick leaded glass windows. Photographic and periscopic examinations were performed less frequently and required transferring specimens to a separate hot cell. Samples were removed for x-ray diffraction analyses. The radiation field from individual fragments was approximately 30 R/h.
Spent fuel fragments from Point Beach (29,292 MWD/MTU burnup), H.B. Robinson (30,000 MWD/MTU), and Shipping port (33,916 MWD/MTU) were tested in 20°C dew point nitrogen cover gas. A modified vacuum oven was used to control the composition of the nitrogen cover gas. The oven was evacuated and flushed six times with nitrogen (deoxidized in 400°C copper fines and bubbled through 20°C distilled water) prior to heating. A 1-psig check valve prevented overpressurization of the vacuum oven during heating. The maintenance of 1-psig cover gas pressure during the exposure at 275°C confirmed the integrity of the oven seals. The oven was cooled to less than 100°C prior to specimen removal to prevent oxidation in atmospheric air. Typical chemical analyses of the nitrogen cover gas before and after exposures at 275°C are given in Tables 2 and 3.

4. EXPERIMENTAL RESULTS

Experimental results are described for non-irradiated pellets and irradiated Point Beach spent fuel.

4.1 NON-IRRADIATED PELLETS

Results of tests conducted in oxidizing and moist nitrogen cover gases are discussed.

4.1.1 Tests in Oxidizing Cover Gases

Initial tests were started at 345°C and 300°C with a continuous weighing system (White et al. 1983); however, the oxidation rates were so rapid that lower temperature tests were required for
evaluating extended storage conditions (see Figure 1). Furthermore, the continuous weighir· system prevented inspection of the specimens for powder formation and limited the number of specimens that could be tested.

The next set of tests was performed between $150^\circ C$ and $250^\circ C$ with interrupted weight measurements. The data indicated variability up to $+100\%$ to $-50\%$ of the nominal (White et al. 1983), even when seemingly similar pellets were tested side-by-side. It was postulated that variability in pellet densities and surface conditions could cause these differences. Therefore, correlations in oxidation behavior were made with differences in pellet character.

A number of variables contributed to the spread in the non-irradiated UO$_2$ data. One of these, ambient air aging (storage time in ambient air after hydrogen anneal or reduction and prior to testing), had a strong influence on the oxidation behavior. The effect of aging time on oxidation behavior at $200^\circ C$ is shown in Figure 2. The effect of aging time on powder induction time is shown in Figure 3. The variables that have been investigated in oxidation tests with non-irradiated UO$_2$ and the impacts of the results are presented in Table 4.

4.1.2 Tests in Moist Nitrogen Cover Gas

Examinations were conducted at 295 to 1165 h of cumulative exposure of non-irradiated UO$_2$ pellets in $20^\circ C$ dew point nitrogen at $225^\circ C$, $250^\circ C$, and $275^\circ C$ in a gamma field of 200,000 R/h. In all cases, the weight gain was less than the detectability limit of the balance (less than 0.1 mg) (see Table 2). No changes in physical appearance could be detected by visual examination. Benefits derived from the nitrogen cover gas were striking since air inleakage during test apparatus development led to significant pellet weight increases and powder formation in less than 1 week at $225^\circ C$.
4.2 IRRADIATED FUEL

Results from atmospheric air tests on Point Beach spent fuel fragments at 230°C and 200°C are compared with results for non-irradiated pellets and fragments in Figures 4 and 5. After 2337 h at 230°C and after 3000 h at 200°C, powder appeared to form in the spent fuel samples, based on sightings through lead glass windows. Closer inspection through a periscope revealed fine fragments with sizes ranging up to 1 mm (see Figure 6). Prior to this time, the fragment size had ranged from 5 to 7 mm. No x-ray analyses have been performed on these specimens. Spent fuel fragments prior to the oxidation test and after 975 h at 230°C are shown in Figure 7; finer fragments had not developed. An inspection at 2763 h indicated that no particulates had yet formed at 200°C. Powder formed in non-irradiated specimens after approximately 230 to 600 h at 230°C, depending on the age of the pellet, and prior to 800 h in the furnaces used for the spent fuel tests.

5. DISCUSSION

During storage of spent LWR fuel in air or a gas containing oxidizing reactants, it is possible that fuel with reactor-breached cladding could oxidize and that the breach could propagate due to the associated fuel swelling resulting from increased lattice parameter and powder formation. Breached spent fuel may not be placed into dry storage intentionally; but non-destructive testing methods cannot assure that all breached rods will be detected, even if the fuel is inspected prior to dry storage. There is a small but real potential for breaches to occur during dry storage. Only 1 rod out of over 5600 monitored during dry storage tests and demonstrations has breached during storage (Johnson and Gilbert 1983b). Even though the amount of reactor-induced breached fuel is small (less than 1%), the release of particulate spent fuel into a storage system could increase the complexity of subsequent fuel handling.
and storage, disposal, or reprocessing operations. Spent fuel oxidation and the implications on the design of dry storage systems are discussed in the following sections.

5.1 OXIDATION BEHAVIOR OF NON-IRRADIATED UO₂

The cost for establishing the effect of potential test variables on spent fuel oxidation behavior would be prohibitively high. Therefore, non-irradiated UO₂ tests were conducted at PNL to determine the importance of variables such as temperature, gadolinia content, radiolysis products in the air cover gas, surface condition of the pellet, and moisture content of the air cover gas on oxidation behavior. Studies in the United Kingdom have identified that fuel density also has an effect (Simpson and Wood 1983).

The effects of several variables on the oxidation behavior of non-irradiated UO₂ were found to be significant. The results were used for guidance in selecting test conditions to optimize the effectiveness of limited tests with spent fuel. A summary of the results of oxidation tests on non-irradiated UO₂ pellets and their impact on spent fuel projects is shown in Table 4. Base on this comparison, tests are being initiated on spent LWR fuel with a range of burnups at temperatures from 135 to 200°C to develop design data.

5.1.1 Phase Diagrams

The U/O system forms a large number of UOₓOᵧ compositions. O/U ratios relevant to LWR spent fuel storage range from 2 to 3 and increase due to oxygen reaction at the surface of the UO₂ and diffusion into the crystalline lattice (Breitung 1978). Ratios increase first
along short-circuited diffusion paths such as dislocations, grain boundaries, subgrain boundaries, and porosity in polycrystalline UO$_2$ fuel pellets (Wang 1981).

A number of phase diagrams have been published (Levin et al. 1964; Levin and McMurdie 1975; Roth et al. 1981); however, these diagrams are not in total agreement. The disagreements are due, in part, to different experimental conditions and differences in interpretation of the results. A typical phase diagram is shown in Figure 8 (Roth et al. 1981). This phase diagram does not show the U$_3$O$_7$ phase that is commonly observed in various forms as a thin film on the surface of UO$_2$ (Aronson et al. 1957). The thickness of the U$_3$O$_7$ film has not been clearly established but has been reported to be limited from 22 nm to less than 100 nm (Smith 1960; Harrison et al. 1967) or to be thicker than 3000 nm (Taylor et al. 1980; Simpson and Wood 1983). The thicker films are cracked because the high-density U$_3$O$_7$ (11.4 g/cc versus 10.97 g/cc for alpha cubic UO$_2$) generates a high tensile stress that may not be sustained in thick films (Fromhold 1976).

Because of the high density of U$_3$O$_7$, it neither leads to fuel swelling nor causes mechanical damage to fuel cladding. U$_3$O$_7$ forms at temperatures as low as 66°C (Dominey 1968) and is responsible for nearly half of the weight gain prior to the formation of U$_3$O$_8$.

At temperatures above 300°C, U$_3$O$_8$ forms so rapidly that U$_3$O$_7$ can be observed only if the formation of U$_3$O$_8$ is suppressed by reducing the oxygen pressures to less than 0.01 atm. U$_3$O$_7$ is formed by slight tetragonal shifts in the cubic UO$_2$ lattice as indicated in Figure 9. Prior to the formation of U$_3$O$_7$, U$_4$O$_9$ is formed in the cubic UO$_2$ lattice (Hoekstra et al. 1961). Slight differences in the appearance of the U$_3$O$_7$ structures in x-ray analyses have led to further classifications of alpha, beta, and gamma forms of U$_3$O$_7$. The changes in the crystal structure, the lattice parameter, and the O/U ratio enable phases such as U$_3$O$_7$ to be identified by x-ray diffraction.
$U_3O_8$ is an orthorhombic low-density phase (8.35 g/cc versus 10.97 g/cc for alpha cubic $UO_2$). Its formation is associated with a volume expansion of at least 24%. Swelling greater than 24% is possible due to the mismatch between grains when $U_3O_8$ at the grain boundaries forces grain decohesion. The surface of a $UO_2$ pellet after some grains were expelled by the formation of $U_3O_8$ in the grain boundaries is shown in Figure 10. The chemical stress associated with fuel swelling from $U_3O_8$ formation can produce mechanical damage to fuel cladding (Bodine et al. 1961; Boase and Vandergraaf 1977; Johnson et al. 1983; Einziger and Cook 1983; Einziger et al. (1984).

According to Figure 8, $U_3O_8$ forms when the U/O ratio approaches 2.67. The development of the various phases as the U/O ratio increases with time for non-irradiated $UO_2$ pellets and Point Beach spent fuel is shown in Figures 11 and 12. The U/O ratio measured by x-ray diffraction of pellet and fragment surfaces increases earlier in non-irradiated $UO_2$ than in spent fuel.

$UO_3$ densities range from 5 to 8.86 g/cc depending on the form. Five forms of $UO_3$ have been described by Hoekstra and Siegel (1961), and at least four others have been reported. $UO_3$ has been found in oxidation product fuel when a strong oxidant such as $NO_2$ was present in the cover gas (White et al. 1983).

5.1.2 Thermodynamics of Uranium-Oxygen Compounds

Numerous studies have been reported on the thermodynamics and phase diagrams for uranium-oxygen compounds (Samsonov 1973; Barin and Knacke 1973; Barin et al. 1977; Stull and Prophet 1971; Wick and Block 1963; and Wagman et al. 1982). The U/O ratio increases from 2 to 3 in an oxidizing environment with progressively lower free energies of formation. Key reactions and the corresponding free energies of reaction are given in Table 5. The negative free energies of reaction indicate
that all of the reactions that increase the O/U ratio are energetically favorable. These reactions are all possible in dry storage regimes, given the reactants and sufficient time.

Since irradiated UO$_2$ contains numerous fission products as impurities, its thermodynamic and chemical properties may be significantly different from non-irradiated UO$_2$ (Aratono and Tachikawa 1978). Fission products that form precipitates compete for oxygen. Fission products that occupy lattice and interstitial sites within the lattice alter the potential for oxidation of the UO$_2$ through defects generated to maintain charge neutrality as well as by changing the valence state of uranium.

Simplified fission reactions to produce U$^{+6}$ or oxygen vacancies follow:

\[ 2U^{+4} + 4O^{-2} \rightarrow U^{+6} + 2M^{+1} + 4O^{-2} \]

for the development of U$^{+6}$ or

\[ U^{+4} + 2O^{-2} \rightarrow 2M^{+1} + O^{-2} + Ov \]

for the generation of oxygen vacancies, Ov, where M and O are fission atoms and oxygen, respectively. An additional factor augmenting the effect of fission products is the concentration of +3 valence plutonium at the surface of the fuel where the oxidation reactions and diffusion are important (Carlsen and Sah 1981). As in the case of non-irradiated fuel with low valence Gd, these oxygen vacancy sites render the structure highly susceptible to oxygen diffusion in an oxidizing atmosphere. The susceptibility of spent fuel to rapid weight gain by oxygen accumulation is shown in Figures 4 and 5.

A number of other reactions have been postulated for fission products with valence states greater than one. The ionic radii of some fission products and postulated defects are shown in Table 6.
The increase in fission product concentration with $UO_2$ burnup is shown in Figure 13. Concentrations of fission products in LWR fuel at discharge and 10 years after discharge are shown in Table 7. The corresponding effect on lattice parameter depends on whether the fuel remains stoichiometric and $U^{+6}$ forms or whether the stoichiometry changes and oxygen vacancies form to maintain charge neutrality. These possibilities are illustrated in Figures 14 and 15 for $UO_2$ containing only gadolinia and for spent fuel. The corresponding data for the non-irradiated urania-gadolinia and the Point Beach as-received spent fuel are plotted on Figures 14 and 15. It appears that the uranium-gadolinia and Point Beach spent fuel follow neither model precisely and may contain a combination of defect types.

5.1.3 Kinetics of $UO_2$ Oxidation

Weight gain during oxidation is due to diffusion of oxygen into surface and short-circuited paths such as grain boundaries, sub-grain boundaries, dislocations, and open porosity. The oxidation kinetics are nearly linear (White et al. 1983), indicating surface reaction rather than diffusion control as assumed by Wheeler (1983). When the $O/U$ ratio approaches 2.67, $U_3O_8$ powder forms. The pressure of the $U_3O_8$ powder within grain boundaries causes grain boundary decohesion and separation of grains, and the expulsion of grains creates fresh $UO_2$ surfaces. Following powder formation, the oxidation kinetics shows a transition to higher rates corresponding to the increased number of $UO_2$ surface sites available for reaction with oxygen. The size of the powder formed by oxidation increases with increasing grain size (Kinugasa et al. 1982).
5.2 OXIDATION BEHAVIOR OF IRRADIATED FUEL

5.2.1 Defect Structure

X-ray analysis of Point Beach spent fuel fragments (flat faces roughly 5 to 6 mm square) removed after 965 h from the 230°C atmospheric air oven and after 1422 h from the 200°C oven revealed weak, broad lines that were characteristic of an oxygen-rich tetragonal \( \text{UO}_2 \) phase. Even though the O/U ratio was 2.32, the diffraction pattern was not characteristic of \( \text{U}_3 \text{O}_7 \). These results are consistent with the observation that no \( \text{U}_3 \text{O}_8 \) powder had formed in these specimens. There was no evidence of an underlying substrate. After 2362 h at 230°C, fine fragments less than 1 mm on edge began to form. Similar fragments also began to form after 3000 h at 200°C and appeared to be very similar to fine fragments that formed from 5 and 10 wt% gadolinia \( \text{UO}_2 \) (Table 4). X-ray diffraction of the fragments that formed in the \( \text{UO}_2 \) containing gadolinia revealed the presence of \( \text{U}_3 \text{O}_8 \).

5.2.2 Effect of Irradiation

The experimental results from oxidation of spent fuel fragments show accelerated oxidation (weight gain) compared with non-irradiated pellets (even after correcting for the geometric surface area/volume ratio). The induction time for powder formation is burnup dependent. These observations led to postulation that irradiation affected the reaction rate of oxygen with the \( \text{UO}_2 \) surface (irradiation-enhanced surface reaction) and that burnup affected oxygen diffusivity (burnup-enhanced oxygen diffusion).
5.2.2.1 Irradiation-Enhanced Surface Reaction

Oxidation tests on UO$_2$ irradiated under conditions where fuel fragmentation did not occur (no increase in geometric surface area) showed accelerated kinetics of weight gain (Harrison et al. 1967). The rate coefficient for oxidation is shown plotted versus burnup in Figure 16. Radiation increased the oxidation rate coefficient by nearly an order of magnitude at values as low as 0.00012 at.% burnup. The enhancement increased slightly, if at all, with burnup. The area under the oxidation rate versus time curve is also insensitive to burnup, indicating that the burnup effect is not due to enhanced diffusion.

The hypothesis is offered that irradiation increases the surface reactivity of the UO$_2$ for oxidation. The effectiveness of irradiation to enhance surface reactivity is reflected in the high initial rate for spent fuel shown in Figures 4 and 5. The insensitivity of the irradiation-enhanced surface reaction rate to burnup is also reflected in the close agreement between Point Beach (29,292 MWD/MTU burnup) PWR spent fuel fragment and CANDU (8,000 MWD/MTU burnup) fuel fragment results at 230°C and 200°C (Figure 17). Strong oxidants have a similar effect on increasing the initial rate for oxidation (White et al. 1983).

5.2.2.2 Burnup-Enhanced Oxygen Diffusion

A useful insight to the effect of burnup on oxidation behavior of spent fuel can be developed from the effect of Gd$_2$O$_3$ on non-irradiated UO$_2$ pellet oxidation behavior. Gadolinia enhances oxidation of non-irradiated UO$_2$ pellets. The enhancement is not due to increased surface reaction since the initial rate is in agreement with that for non-irradiated pellets with no gadolinia. The oxidation rate increase occurs after considerable oxidation and is attributed to increased oxygen diffusion caused by increased grain boundary misfit energy associated with the defects induced by gadolinia (Catlow 1979; Ohmichi et al. 1981; Une and Oguma 1983; Grossman et al. 1972) and
gadolinia effects on the chemical oxidation potential. Doping $\text{UO}_2$ with simulated fission products (fissia) has also been found to alter the oxidation behavior (Smith 1960). These results demonstrate that oxidation kinetics are affected by low valence fission products.

The burnup-enhanced diffusivity concept leads to the prediction that burnup affects the oxidation behavior of spent fuel. Higher burnup fuels contain a higher concentration of fission products with valence below $U^{+4}$ and thereby increase the number of $U^{+6}$ ions and oxygen vacancy sites and lattice distortions to enhance oxygen diffusivity.

Oxidation results for spent BWR fuel with a burnup of 17,200 MWd/MTU (Johnson et al. 1983) showed that less than half as much $\text{U}_3\text{O}_8$ formed as was predicted, based on conversion of stoichiometric $\text{UO}_2$ to $\text{U}_3\text{O}_8$. This fuel contained ~3 wt% gadolinia. The rate of cladding crack propagation due to $\text{U}_3\text{O}_8$ formation was an order of magnitude lower than for non-gadolinia-bearing spent fuel (Figure 18). The 3 wt% gadolinia apparently caused the fuel to behave as if it contained more fission products (higher burnup) than actually produced by irradiation.

The increase in the time to induce $\text{U}_3\text{O}_8$ and the increasing levels of weight gain with burnup are associated with the $O/U$ concentration gradient as controlled by the burnup-enhanced diffusivity. As illustrated in Figure 19, a large $O/U$ concentration gradient develops at the surface of the low diffusivity non-irradiated $\text{UO}_2$; consequently, the $U/O$ level of 2.67 for $\text{U}_3\text{O}_8$ formation is readily attained. Spallation occurs by grain expulsion from the surface (Figure 10). Longer times are required to build up an $O/U$ ratio of 2.67 at the surface of $\text{UO}_2$ with burnup-enhanced diffusivity; hence, $\text{U}_3\text{O}_8$ formation is associated with larger levels of weight gain and longer times.

The effect of fission products and hence burnup to delay the time for spallation is reflected in the results of the NRC test on BWR spent fuel from Peach Bottom-II with an average bundle burnup of 69650
13,000 MWd/MTU (Einziger and Cook 1983; Einziger and Cook 1984). The effects of \( \text{U}_3\text{O}_8 \) formation was first detected at a hole drilled near the end of a rod during examination at 2235 h. The burnup level and the fission product concentration were lowest at the end where cladding strain was first observed. \( \text{U}_3\text{O}_8 \) formation at the center of the rod, where burnup and fission products were higher, required nearly 5962 h of testing. The rapid rate of cladding splitting at the end of the rod was attributed to the large surface area for reaction between \( \text{UO}_2 \) and oxygen since the drilled hole was located at a pellet end interface. The hole at the center of the rod was not near a pellet end interface, but this probably does not account for the much longer time required for \( \text{U}_3\text{O}_8 \) formation since the powder induction time was not sensitive to surface area in oxidation tests on non-irradiated \( \text{UO}_2 \) pellets and fragments (Table 4).

5.2.3 Cladding Breach Size Effects

It has been anticipated (Einziger and Cook 1983) that small cladding breaches may tend to become plugged by cladding oxidation at the breach and by extrusion of swelling fuel particulates into the breach orifice. However, Zircaloy oxidation is very slow at temperatures below 400°C (Johnson and Gilbert 1983a, 1983b; Stokes and Blankenship 1983); and diffusion of oxygen through a porous particulate fuel plug is inevitable, given the time available during interim storage.

To study the oxidation of spent fuel enclosed in cladding, it is necessary to select fuel rods with reactor-induced breaches (Johnson et al. 1983) or to perforate the cladding. Methods of perforation have ranged from inducing pinhole stress corrosion cracks (Einziger et al. 1984) to drilling holes through the cladding (Novak and Hastings 1983; Novak et al. 1983; Einziger and Cook 1983; Einziger et al. 1984) or cutting slits (Boase and Vandergraaf 1977). The spent fuel rods were heated, exposed to air, and then visually inspected and measured for
transverse growth. In some cases, fission gas release during cladding penetration was used to verify complete penetration (Einziger and Cook 1983).

Results from oxidation tests on full-length CANDU spent fuel rods (Boase and Vandergraaf 1977) showed that the quantity of fuel reacted depended on the size of the cladding defects. The tests were conducted for up to 100 h at 300°C. On a time scale, the cladding provided protection of a factor of approximately three for a defect diameter of 0.5 mm compared with a longitudinal slot 25 mm long by 1.5 mm wide. Therefore, a small stress corrosion crack with a diameter of about one grain may provide even greater protection than a factor of three compared with bare pellet fragments. However, at extended storage times and temperatures, the kinetics changes from supply limited to diffusion limited and the cladding is expected to become less protective in the vicinity of the cladding breach. However, significant cladding protection is provided for fuel away from the breach site.

5.2.4 ISFSI Performance

The CEX test included loading defective bundles of Bruce and Pickering spent fuel into dry and moist air cover gases at 150°C (Ohta 1980). Since the experiment was loaded in dry air in 1980 and moist air in 1981, a significant time base is accumulating. Examination of the fuel after 3 or 4 years indicating that no measurable degradation has occurred would provide a strong supportive data base for storing LWR spent fuel at temperatures up to 150°C. Spent fuel initially placed into storage at 150°C would remain at this temperature for only a short time and would cool due to declining decay heat. Furthermore, the fuel burnup in the CEX test is at the level where induction times for powder formation are likely to be lower than in spent LWR fuel.
Data generated in testing programs may not include all of the effects that exist in a dry storage installation. The tests have been performed in temperature-controlled furnaces to approximately 2°C within the nominal control temperature. Under non-isothermal conditions in storage systems, fuel temperatures will be subject to nocturnal changes in external ambient temperatures, seasonal fluctuations, and gradual cooling due to declining decay heat. The time-dependent temperature fluctuations will produce reversible ventilation through cladding breaches and thereby cause more oxygen to be introduced to the fuel than in a highly controlled isothermal test.

Allowable initial maximum storage temperatures were estimated as a function of spent fuel age since reactor discharge and burnup (Figure 20) using cumulative damage temperature-time methodology (Gilbert et al. 1983; White et al. 1983; Wheeler 1983). The analysis used the storage system cooling curves described in Figure 21. The allowable storage temperature depends on the burnup and the age of the spent fuel after reactor discharge and may be lower for burnup levels below approximately 15,000 MWd/MTU because of the effect of burnup on time for induction of $\text{U}_3\text{O}_8$ (Figure 22). The analysis for Figure 20 assumed a low burnup and a fuel highly susceptible to early $\text{U}_3\text{O}_8$ formation. At burnup levels above 15,000 MWd/MTU, the burnup-enhanced oxygen diffusion reduces the oxygen concentration gradient at the surface (Figure 19) and thereby increases the time before $\text{U}_3\text{O}_8$ forms (Figure 23). The ends of LWR fuel rods may be more susceptible to $\text{U}_3\text{O}_8$ formation than the rest of the fuel due to lower burnup levels. However, the ends will be stored at lower temperatures in most dry storage concepts.

Figure 20 was computed with a value of 0.08 for alpha (the ratio of the induction times for $\text{U}_3\text{O}_8$ formation for irradiated to non-irradiated $\text{UO}_2$), which represents a conservative case for formation of $\text{U}_3\text{O}_8$ in spent fuel. As illustrated in Figure 22, higher burnups are expected to increase the induction times and allowable initial temperatures. However, the delay in $\text{U}_3\text{O}_8$ formation with increasing
burnup may not be as beneficial below 200°C as at temperatures between 200°C and 283°C due to the temperature dependence of oxygen diffusivity.

6. CONCLUSIONS

- Important variables that affect the oxidation behavior of UO₂ include temperature, aging time, surface area, gamma field, NO₂ in cover gas, gadolinia content, density, and moisture.

- Non-irradiated UO₂ or spent fuel with burnup levels below 15,000 MWD/MTU is more susceptible to U₃O₈ formation than LWR fuel with burnup levels above 15,000 MWD/MTU.

- Favorable results of the Canadian CEX test in the burnup range where spent fuel readily oxidizes will support a temperature limit of at least 150°C. Storage above 150°C may require selection of a fuel with adequate burnup and age since reactor discharge.

- Analysis and extrapolation of data on the oxidation behavior of non-irradiated UO₂ and spent fuel suggest that initial dry storage temperatures greater than 150°C will be acceptable for LWR fuel. Spent fuel tests between 150°C and 200°C are required to establish dry storage design temperatures above 150°C.

- Future studies on spent fuel oxidation should include gamma radiation in a static closed cover gas. Open systems do not allow accumulation of radiolysis products.
Tests between $150^\circ$C and $230^\circ$C on spent LWR fuel are being initiated under the DOE CSFM Program to develop design data and to verify the methodology for computing allowable initial storage temperatures.

Nitrogen is an acceptable dry storage cover gas if it is reasonably dry.

ACKNOWLEDGMENTS

The contributions of the following PNL personnel are acknowledged: R.E. Thornhill for hot cell experiments; H.E. Kissinger for x-ray diffraction results; J.E. Coleman for scanning electron microscopy; N.J. Wildung for computer graphics; L.S. Dake for XPS analysis of surface chemical compositions; R.J. Traub for measurement of gamma flux by the thermoluminescent dosimeter method; and S.K. Edler for technical editing. Appreciation is expressed to the TVA for permission to show unpublished TVA data on archive non-irradiated UO$_2$ pellets. Acknowledgments are gratefully made to Dr. J.L. Bates, Dr. J.L. Ryan, C.W. Griffin, Dr. A.B. Johnson, Jr., D.R. Oden, and Dr. R. Wang of PNL and Dr. R.A. Busch of Exxon Nuclear Company, Inc., for technical discussions and critical reviews of the manuscript.

REFERENCES


### TABLE 1:
PARTICLE SIZE DISTRIBUTION FOR SPENT FUEL SPECIMENS

<table>
<thead>
<tr>
<th>Mesh Size</th>
<th>H. B. Robinson</th>
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\(^{(a)}\) 3A - H. B. Robinson spent fuel
3B - nonirradiated pellets
3C - Point Beach spent fuel
3D - Shippingport spent fuel.
TABLE 4: IMPACTS OF OXIDATION STUDIES WITH NONIRRADIATED UO$_2$ ON SPENT FUEL PROJECTS

<table>
<thead>
<tr>
<th>Variable</th>
<th>Nonirradiated UO$_2$ Results</th>
<th>Impact on Spent Fuel Studies</th>
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<tr>
<td>Temperature</td>
<td>Oxidation proceeds rapidly above 230°C.</td>
<td>Focus spent fuel studies below 230°C.</td>
</tr>
<tr>
<td>Aging</td>
<td>Aging in ambient air reduces oxidation rate.</td>
<td>Examine effect of aging.</td>
</tr>
<tr>
<td>Surface area</td>
<td>Fragmentation increases rate approximately 5 to 7 times and has minor effect on particulate induction time.</td>
<td>Measure fragment size distribution.</td>
</tr>
<tr>
<td>Gamma field</td>
<td>Gamma field enhances oxidation.</td>
<td>Impose gamma field spent fuel tests.</td>
</tr>
<tr>
<td>NO$_2$ in cover gas</td>
<td>Simulated NO$_2$ radiolysis product enhances oxidation.</td>
<td>Impose gamma field fuel spent tests.</td>
</tr>
<tr>
<td>Gadolinia</td>
<td>Gadolinia delays formation of U$_3$O$_8$ and results in the formation of fine fragments.</td>
<td>Include gadolinia spent fuel in tests; provide insight on fission product effects.</td>
</tr>
<tr>
<td>Density</td>
<td>Minor effect for closed porosity above 94.5% TD.</td>
<td>Measure density of spent fuel specimens.</td>
</tr>
<tr>
<td>Moisture</td>
<td>Moisture slightly increases oxidation rate.</td>
<td>Perform spent fuel tests under typical dry storage moisture conditions.</td>
</tr>
<tr>
<td>Temperature changes</td>
<td>Oxidation response to temperature changes supports cumulative damage models.</td>
<td>Perform temperature change tests on spent fuel.</td>
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<tr>
<td>Thermal cycles</td>
<td>Minor effect from thermal cycles.</td>
<td>Confirm slight thermal cycle effect for spent fuel.</td>
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### Table 5: Thermodynamics of Chemical Reactions of Oxides of Uranium with Oxygen at 200°C

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<tr>
<th>Reactions and Products (kcal/mole)</th>
<th>Free Energy of Reaction (kcal/mole)</th>
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<tr>
<td>(4\text{UO}_2 + \frac{1}{2} \text{O}_2 \rightarrow \text{U}_4\text{O}_9) &lt;br&gt;((-1076) + (-12) -1120)</td>
<td>(-32) (Barin and Knacke 1973)</td>
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<tr>
<td>(3\text{UO}_2 + \frac{1}{2} \text{O}_2 \rightarrow \text{U}_3\text{O}_7) &lt;br&gt;((-740) + (0) -775)</td>
<td>(-35) (25°C)(^{(a)}) (Wagman et al. 1982)</td>
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<tr>
<td>(3\text{UO}_2 + \text{O}_2 \rightarrow \text{U}_3\text{O}_8) &lt;br&gt;((-807) + (-24) -890)</td>
<td>(-86) (Barin and Knacke 1973)</td>
</tr>
<tr>
<td>((-718) + (0) -775)</td>
<td>(-57) (Samsonov 1973)</td>
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<tr>
<td>(\text{U}_3\text{O}_7 + \text{O}_2 \rightarrow \text{U}_3\text{O}_8)</td>
<td>(-30) (25°C)(^{(a)}) (Wagman et al. 1982)</td>
</tr>
<tr>
<td>((-775) + 0 -805)</td>
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<tr>
<td>(\text{U}_3\text{O}_8 + \frac{1}{2} \text{O}_2 + \text{3UO}_3) &lt;br&gt;((-890) + (-12) -915)</td>
<td>(-13) (Barin and Knacke 1973)</td>
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<td>((-775) + (0) -789)</td>
<td>(-14) (Samsonov 1973)</td>
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\(^{(a)}\) Data not available at 200°C.
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<th>Defect</th>
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<td>fission product</td>
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<td>Oxygen interstitial vacancy;</td>
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<td>interstitial complex defects</td>
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Total grams from fission products = $3.70 \times 10^{4}$ g/tU

Total grams from actinides = $9.63 \times 10^{5}$ g/tU $1.00 \times 10^{6}$ g/tU

(a) Croff (1980).
Figure 1: Continuous weight gain data of non-oxidized 02 pellets at 200°C.

Figure 2: Effect of ambient air aging on oxidation of 02 pellets at 200°C.

Figure 3: Effect of ambient air aging on powder induction in non-oxidized 02 pellets at 200°C.

Figure 4: Comparison of oxidation of spent fuel with non-oxidized pellets at 230°C.
FIGURE 5: Oxidation Behavior of Point Beach PWR Spent Fuel at 200°C (~29,000 MWD/MTU burnup)

FIGURE 6: Fragmented Point Beach Spent Fuel

FIGURE 7: Point Beach Spent Fuel Prior to Oxidation Test

FIGURE 8: Phase Diagram for UO₂ to UO₃ (Roth et al. 1981)
FIGURE 9: Effect of O/U Ratio on Lattice Parameter and Tetragonality of $U_3O_7$ (Pearson 1958)

FIGURE 10: Surface Spallation from $U_3O_8$ Formation in Nonirradiated $UO_2$ Pellet

FIGURE 11: Comparison of Surface Oxygen Concentration and Time Results for Nonirradiated $UO_2$ Pellets and Point Beach Spent Fuel Fragments at 230°C

FIGURE 12: Comparison of Surface Oxygen Concentration and Time Results for Nonirradiated $UO_2$ Pellets and Point Beach Spent Fuel Fragments at 200°C
**FIGURE 13:** Fission Product Concentration Versus UO₂ Burnup

**FIGURE 14:** Effect of Gadolinia on Lattice Parameter and O/M Ratio for Nonirradiated Urania-Gadolinia Pellets (Ohmichi et al. 1981)
**Figure 15:** Effect of burnup on lattice parameter and O/M ratio for irradiated UO₂ (Davies and Ewart 1971) and burnup simulated fuel (Une and Oguma 1983).

**Figure 16:** Effect of burnup on oxidation kinetics at 320°C (Harrison et al. 1967).

**Figure 17:** Comparison of Point Beach PWR spent fuel (~29,000 MWD/MTU) and CANDU spent fuel (8,000 MWD/MTU) oxidation behavior in air.
FIGURE 18: Effect of Gadolinia on Oxide Front Velocity

FIGURE 19: Effect of Burnup on O/U Profile Prior to $U_3O_8$ Formation in Center of LWR Fuel

FIGURE 20: Maximum Temperature to Prevent $U_3O_8$ Formation in Irradiated $UO_2$ Under Cooling Conditions Versus Age of Fuel at Insertion

FIGURE 21: Temperature History for Spent Fuel in PWR Storage Cask ($T = $temperature at time $t$; $T_o = $initial fuel temperature; $T_1 = $fuel temperature in dry storage 1 year after reactor discharge)
FIGURE 22: Effect of Spent Fuel Burnup on Induction Time for U₃O₈ Powder Formation

FIGURE 23: Effect of Burnup on U₃O₈ Powder Induction at a Specific Temperature Between 200°C and 230°C
APPENDIX A

Activities are being conducted internationally on the oxidation of UO₂ that define the conditions under which spent fuel can be preserved during interim dry storage without compromising retrievability or acceptability for future monitored retrievable storage facilities or disposal in a repository. The range of conditions and parameters and highlights of results from these studies are described in Table A-1.

Although the studies are not complete, the results indicate that the storage of spent fuel in oxidizing cover gas should not exceed a yet-to-be-defined temperature-time exposure, which is expected to be associated with an initial storage temperature of 150 to 200°C. The allowable initial storage temperature will depend on the installation design and the age and burnup of the fuel to be stored. Fuels that have been in pool storage for several years will cool at slower rates during dry storage than fuels recently removed from the reactor. Determination of the specific temperature-time exposure to prevent U₃O₈ powder formation is not yet identified and requires longer term data from tests that are now under way or being planned.
# TABLE A-1:
ACTIVITIES DIRECTED TOWARD DETERMINATION OF
SPENT UO₂ FUEL OXIDATION PROPERTIES

<table>
<thead>
<tr>
<th>Investigator</th>
<th>Test Conditions</th>
<th>Highlights</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>NONIRRADIATED UO₂ PELLETS</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>CEGB/AERE**(a)**</td>
<td>175 to 250°C</td>
<td>Temperature dependence.</td>
</tr>
<tr>
<td></td>
<td>Density</td>
<td>Oxidation inversely related to density.</td>
</tr>
<tr>
<td>TVA</td>
<td>140 to 300°C</td>
<td>Rapid powder formation above 252°C.</td>
</tr>
<tr>
<td></td>
<td>Nonirradiated pellets</td>
<td>Data in nominal agreement with PNL results for pellets aged in ambient air.</td>
</tr>
<tr>
<td>DOE/PNL**(b)**</td>
<td>150 to 350°C</td>
<td>Rapid oxidation above 200°C.</td>
</tr>
<tr>
<td></td>
<td>Variable moisture</td>
<td>Moisture enhances oxidation.</td>
</tr>
<tr>
<td></td>
<td>Gamma radiation field</td>
<td>Gamma radiation accelerates oxidation and U₃O₈ formation in a closed system due to buildup in radiolysis products. UO₃ can form.</td>
</tr>
<tr>
<td></td>
<td>UO₂ composition</td>
<td>Gd₂O₃ accelerates oxidation and delays U₃O₈ formation.</td>
</tr>
<tr>
<td></td>
<td>Ambient aging in air</td>
<td>Retards oxidation with protective UO₃ surface.</td>
</tr>
<tr>
<td></td>
<td>after sintering</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Thermal cycling</td>
<td>No effect.</td>
</tr>
<tr>
<td><strong>IRRADIATED UO₂</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Spent fuel fragments</td>
<td></td>
<td>Spent fuel oxidizes faster than nonirradiated pellets due to fragmentation and increased concentration of oxygen vacancies, but U₃O₈ formation is delayed at high burnup.</td>
</tr>
<tr>
<td>Investigator</td>
<td>Test Conditions</td>
<td>Highlights</td>
</tr>
<tr>
<td>----------------------</td>
<td>----------------</td>
<td>-----------------------------------------------------------------------------</td>
</tr>
<tr>
<td>AECL/CRNL/OH(c)</td>
<td>150 to 325°C</td>
<td>Significant oxidation at 175°C and above.</td>
</tr>
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<td></td>
<td>Spent fuel variables</td>
<td>Fission products accelerate weight gain but delay the onset of $U_3O_8$ formation.</td>
</tr>
<tr>
<td></td>
<td>Spent fuel fragments</td>
<td>Spent fuel oxidizes faster than nonirradiated $UO_2$.</td>
</tr>
<tr>
<td></td>
<td>PHWR spent fuel rods</td>
<td>Oxidation increases with increased number of holes in cladding.</td>
</tr>
<tr>
<td></td>
<td>Less than 800 h</td>
<td>No powder below 250°C.</td>
</tr>
<tr>
<td>NRC/HEDL/EG&amp;G(d)</td>
<td>230°C</td>
<td>BWR cladding crack extension in air.</td>
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<td></td>
<td>Defective PWR and BWR spent fuel rods</td>
<td>$U_3O_8$ powder formed in BWR rods in less than 3 months.</td>
</tr>
<tr>
<td>EPRI/TVA/HEDL/ANL(e)</td>
<td>250 to 350°C</td>
<td>Spent PWR fuel rods.</td>
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<tr>
<td></td>
<td>Defective PWR spent fuel rods</td>
<td>Variation in cladding breach size.</td>
</tr>
<tr>
<td>CEGB/AERE(a)</td>
<td>Irradiated HTGR fuel specimen in 25 vppm moisture</td>
<td>Slight reduction of $U_3O_8$ induction time.</td>
</tr>
</tbody>
</table>

(a) Simpson and Wood (1983).
(b) White et al. (1983).
(c) Boase and Vandergraaf (1977); Hastings and Novak (1983); Novak et al. (1983).
(e) Einziger et al. (1984); Einziger and Strain (1984).
EFFECT OF CLADDING DEFECT SIZE ON THE OXIDATION OF IRRADIATED SPENT LWR FUEL BELOW 360°C

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Westinghouse Hanford Company

and

R.V. Strain
Argonne National Laboratory

ABSTRACT

Tests on spent fuel fragments and rod segments were conducted between 250 and 360°C to relate temperature, defect size, and fuel oxidation rate with time-to-cladding-splitting. Defect sizes from 760 μm diameter down to 8 μm, the size of an SCC type breach, were used. The time to spallation for the bare fuel fragments could be described by \( I(\text{hrs}) = 2 \times 10^{-6} \exp\left(\frac{194 \text{ kJ/mol}}{RT}\right) \). Above 283°C, the time-to-cladding-splitting was longer for the smaller defects. The enhancement of the incubation time by smaller defects steadily decreased with temperature and was not detected at 250°C.
1. INTRODUCTION

Knowledge of the oxidation behaviour of spent fuel is important in order to minimize the atmospheric monitoring of a dry storage system and lengthen the acceptable response time to accident scenarios. Approximately 0.01 percent of the LWR rods discharged for the reactors contain a breach (Garzarolli et al, 1979). Since there is no apparent practical way to separate pinhole breaches, some of these breached rods may be placed in dry storage. If the fuel is stored or exposed to air and if the $\text{UO}_2$ oxidizes to $\text{U}_3\text{O}_8$, there is a stress placed on the cladding due to the reduction in density from 10.4 g/cc (95 percent dense $\text{UO}_2$) to 8.4 g/cc. This stress can cause the cladding to split open creating contamination within the container. This could be a concern during storage canister unloading.

Early oxidation on $\text{UO}_2$ indicated that below 250°C, $\text{U}_3\text{O}_7$ is the stable oxide and above 250°C, $\text{U}_3\text{O}_8$ is the stable oxide, although there is no indication in the phase diagram that this should be the case (Roth et al, 1981). Only three studies with conflicting results were made on irradiated fuel (Harrimen et al, 1981; Smith, 1980; Schmets, 1963). While recent studies on unirradiated $\text{UO}_2$ pellets (Simpson and Wood, 1982) indicate that $\text{U}_3\text{O}_8$ can form at temperatures as low as 200°C, the effects of the irradiation (Boase and Vandergraaf, 1977; White et al, 1982; Hastings and Novak, 1982) are still not clear. Oxidation studies using artificially (Boase and Vandergraaf, 1977; Einziger and Cook 1984; Novak and Hastings, 1982) or grossly defected rods (Johnson et al, 1982) have indicated that the cladding will strain then split when the fuel oxidizes, but no relationship between the oxidation behaviour of bare fuel fragments and irradiated defected fuel rods has been developed.
Oxidation data may be interpreted in terms of an incubation and a growth period but there are a number of definitions of incubation. For bare fuel particulate, the end of the incubation period can be defined as the time when fuel spallation is first visually observed. End of incubation for fuel rods can be defined as the time when the cladding starts to split open. While these definitions are physically equal, they are functionally similar since both define a time when the contamination potential increases significantly. The present study attempts to determine any enhancement of the incubation time of breached fuel rod cladding containing micron size defects as compared to spent fuel rods with large defects or bare fuel.

2. TEST DESCRIPTION AND MATERIALS

To determine if cladding with a SCC (stress corrosion crack) type breach extends the incubation period relative to bare spent fuel, testing was conducted in a temperature range where bare fuel oxidation would occur in a reasonably short time. A long term test was conducted at 250°C and a series of shorter term tests at 360°C to 280°C. Since the oxidation temperature dependence appears to have an Arrhenius behaviour, the general testing philosophy was to test at higher temperatures where more data could be gathered in a short time, and then extrapolate to the lower dry storage temperatures where the oxidation would be much slower but the storage time much longer.

2.1 SAMPLE PREPARATION

Turkey Point Unit 3 PWR rods with a burnup of 27 GWD/MTu were dry cut into 8-in. samples. Approximately 19 mm of fuel was removed from each end of the samples to allow attachment of pressure leads and/or air
tight caps. The test rods were not characterized, but companion rods from the same assembly had previously (see Table 1) been both nondestructively (Davis, 1980) and destructively examined (Davis and Pasupathi, 1981) for gas release, diametral strain, gamma profile, grain size and cladding condition. Turkey Point fuel is representative of low gas release PWR fuel.

The rod section samples were defected with either a 760 µm diameter drill or by pressurization and low strain rupture. In order to decrease the time to rupture, elemental iodine was included in the defecting atmosphere. The defect range was chosen so the largest defect would approximate the defect size used in CANDU studies (Boase and Vandergraaf, 1977) and the smallest defect was representative of an SCC type crack (Mattas et al, 1979). The size of the defect in terms of an equivalent circular diameter was determined by rod pressurization and ranged from 8 to 37 µm. The two types of defects are illustrated in Figure 1. The diameter of the rod was measured with a micrometer at a number of axial locations before and after defecting. Pertinent characteristics of the test samples are given in Table 2.

2.2 TESTING PROCEDURE

A typical oxidation test consisted of one rod segment containing a 760 µm drilled defect, one or more rod segments containing the defects in the 12-40 µm range and one or more platinum crucibles containing between 27 and 52 fuel fragments, each with an approximate average spherical diameter of ~0.4 cm (See Figure 2). The samples were placed in a vacuum chamber which was continually purged with building air. Air flow meters were located both on the input and output side of the chamber. The tube furnace using two control and two measuring thermocouples, maintained the temperature within ±5°C. Periodically, the test was interrupted and the samples removed for measurement. The crucibles were weighed and the fuel visually checked for any sign of fuel
spallation. The defects in the fuel rod segment were photographed and the rod diameter was measured in two azimuthal directions at the breach site and other axial locations. If the defect had started to extend, the length of the defect was measured.

3. RESULTS AND DISCUSSION

3.1 BAR FUEL OXIDATION

Periodically, the nickel-chromium crucibles of fuel fragments were weighed and observed for fuel spallation. The results are plotted in Figure 3. At 283°C and 250°C, duplicate samples, started at different times, showed excellent repeatability. At 360°C, substantial spallation had already occurred at the initial 4 hour observation. By the second observation, the samples had been reduced to a fine powder (See Figure 4). Spallation appears to require a weight gain between 2.5 and 3 percent indicating a substantial conversion of UO₂ and U₃O₈. There does not appear to be a temperature dependence to the weight gain at spallation. Above 280°C there is a substantial weight gain above 3.96 percent necessary for complete conversion of the UO₂ to U₃O₈. This may be due to water of hydration or further oxidation beyond U₃O₈ toward UO₃. No x-ray diffraction has been done to determine the crystallographic state of the oxide. At 250°C the weight gain appears to level out at ~2.5 percent after ~1100 hours and remain so through spallation at ~5100 hours. This is in contrast to the experience with CANDU irradiated fuel (Hastings and Novak, 1982) at 250°C and Point Beach fuel (White et al, 1982) at 230°C where the weight gain continued to increase even at 3 percent.
The time to spallation as a function of temperature is plotted in Figure 5 along with other spallation studies. The spallation follows an Arrhenius behaviour with an activation energy of 194 ± 24 kJ/mol which is higher than the activation energy of 147 kJ/mol found by White et al and 121 kJ/mol determined by Simpson and Wood for oxidation of unirradiated UO₂. This may not be significant since there is almost a 30 percent difference in activation energies between the two unirradiated studies and the irradiated fuel used in the present study is from neither of those batches. The incubation period for the irradiated fuel is also larger than that of the unirradiated fuel but it is premature to infer an irradiation effect since no oxidation kinetics of unirradiated Turkey Point fuel are available.

Hastings and Novak claim that cladding splitting occurs at 2 percent diametral cladding strain corresponding to 0.6 percent fuel weight gain. Interpolating between the start of the experiment and the first weight measurements, one can obtain an error band on the present data for 0.6 percent weight gain. One such interpolation is shown in Figure 3. When the 0.6 percent weight gain indication period is plotted for both the present and Canadian work, as in Figure 6, there is reasonable agreement.

3.2 DEFECTED ROD BEHAVIOUR

Rod samples were frequently visually examined for the appearance of cracks and measured with a micrometer to determine if there was any cladding strain which might indicate fuel oxidation. After cracks formed, the rate propagation, mode of cladding cracking and dispersal of fuel were recorded.
Incubation, for the purpose of rod behaviour, is being defined as the time when a through-the-wall crack starts to propagate. Since continuous observations were not being made, the incubation time was bracketed by the last observation of unpropagated defect and first observation of a propagated defect. More precisely, the incubation times were obtained by the extrapolation of the crack growth data to zero crack growth. The procedure is illustrated in Figure 7 for 283°C. Note that for the 760 µm defect, that cladding cracking had already started by the first observation and that for the 12 µm defect, the extrapolation of the crack growth data would have given an incubation time much shorter than the last observation of no crack propagation. If the extrapolation indicated an incubation time shorter than the last observation of an unpropagated defect, then the error in the crack growth rate was too large and the last observation of an unpropagated defect was used as the incubation time. Presently there is no crack growth data for the 37 µm defect at 250°C which accounts for the uncertainty in the incubation period. The incubation time and crack growth rates are given in Table 3.

An Arrhenius plot of the "splitting" incubation period is given in Figure 8. The "spallation" incubation period for the companion bare fuel fragments and "splitting" incubation period from other studies are shown for comparison purposes. Except for two rod samples tested at 283°C, all the rod samples and bare fuel fragments came from the central portion of the rod, where the burnup profile was flat and the cladding had crept down on the fuel during irradiation. The other two rod samples, used at 283°C, were from the ends of the rod where the burnup was significantly lower and there was a considerable fuel cladding gap. The majority of the samples indicated that the incubation time varied inversely with the size of the original defect implying that the defect was inhibiting access of O₂ to the fuel. The availability of O₂ to the fuel controls the rate of oxidation, i.e., the fuel is in an O₂ starved condition. However, as the temperature drops below ~250°C, the oxidation kinetics become rate controlling.
Two samples tested at 283°C were taken from the ends of the rod. Based on the "balling performance of bare fuel the defects in these samples propagated earlier than expected. The larger 760 μm defect propagated first indicating the defects still appear to be limiting access of O₂ to the fuel. The lower burnup of these samples may be affecting the rate of oxidation but there is no conclusive evidence that this is the case. Testing is continuing at 295°C to determine the cause of this apparent anomaly.

The strain-to-breath (STB) is in the neighborhood of 6.5 to 7.5 percent for the 760 μm circular defect but <1 percent for the small SCC type defects (see Table 4). The STB for the 760 μm circular defect was in agreement with the 6.5 percent strain necessary to propagate a crack in the Peach Bottom fuel. The difference in STB is not surprising since the pinhole cracks are really not circular but are sharp, thus providing a high stress concentration as well as being oriented longitudinally through the wall, making axial propagation easier. This is supported by the lower stress needed for propagation since 6.5 percent strain corresponds to approximately 85 percent local conversion of the UO₂ to UO₃ for a PWR rod while 1 percent strain is only ~20 percent conversion (Einziger and Cook 1984). Somewhat surprising is that in neither case does there appear to be a temperature dependence to the STB. The cladding is irradiation hardened in the reactor. In the 20 to 60 hours required to breach at 360°C there should have been 60 percent recovery of the damage (Kemper and Zimmerman, 1957) while the cladding should not have shown any recovery by the time the cracks propagated at 325 and 283°C (Einziger and Kohli, 1984). Hardness measurements for clarification of the recovery are planned. In one case at 283°C the split initiated at a location away from the original defect; the strain at the split location was 5.3 percent. If the cladding has not recovered, as is expected, the 1 percent cladding strain-to-breath criteria for intact rods may be overly conservative.
After the split initiates, it propagates in a number of different configurations as shown in Figure 9. Straight propagation up the rod (Figure 9a), spiral cracks (Figure 9b) and radial turns resulting in cladding flaps 270° around cladding (Figure 9c) have all been observed. The splits that developed in the CANDU rods (Boase and Vandergraaf, 1977; Novak and Hastings, 1982) Peach Bottom rod, (Einziger and Cook, 1984) and Point Beach rod (Johnson et al, 1982) were all axial splits. As can be seen from Table 4, there is no correlation between split configuration and either temperature or initial defect shape. No reason can be advanced at this time for the various configurations. While all three types of cracks result in significant dispersal of powdered fuel, the flap type crack is of most concern since it may also jeopardize the structural integrity of the rod.

Once a cladding flap starts to form, it does not appear practical or meaningful to define and measure a crack propagation rate. However, for the six splits that were either axial and spiral in nature, a one directional split velocity was determined. For those splits with flaps, a pre-flap crack growth rate was determined. These are plotted in Figure 10 along with the axial front velocities determined in other tests by Einziger and Cook, Novak and Hastings and Boase and Vandergraaf. There is generally good agreement among the data. The activation energy for the combined data is 86 ± 13 kJ/mol, less than half the activation energy of 194 ± 24 kJ/mol found for the spallation of bare fuel.

4. CONCLUSIONS

Oxidation studies have been conducted on irradiated rod segments containing defects from 8 to 760 μm equivalent diameter and bare fuel fragments. Although the program is continuing, the following conclusions can be drawn:
1. From 250°C to 360°C at least some of the UO$_2$ is oxidizing past U$_3$O$_7$.

2. The geometric configuration of a cladding crack is unpredictable. In some cases there appears to be uniform axial crack propagation; in other cases the crack starts to take a radial turn, then spiral around the cladding.

3. Once the cladding starts to crack, the fuel may maintain its fragment-like appearance but can easily disperse.

Two other preliminary conclusions can be drawn which should be further developed as the 250°C and 295°C tests continue and hardness measurements, x-ray diffraction studies and SEM examinations made. These preliminary conclusions are:

1. Cladding with breaches substantially delay the incubation of fuel oxidation relative to bare fuel. The smaller defects provide the greatest inhibition, which is consistent with the defect limiting the supply of oxygen. This effect appears to decrease as the temperature drops to 250°C.

2. The fuel rod performance may be dependent upon the axial position of the cladding defect due to either burnup or gap size variations.

ACKNOWLEDGEMENTS

Frank Yaggee's (ANL) ability to put SCC type defects in spent fuel cladding was a key to success of this project. Frank's contribution is appreciated by the investigators. The experimental work was conducted by the staff of Argonne's alpha-gamma hot cell facility. The authors
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Davis, R.B. and Pasupathi, V., 1981, Data Summary Report for the Destructive Examination of Rods G7, G9, J8, 19, and H6 from Turkey Point Fuel Assembly B17, HEDL-TME 80-85.


Roth, R.S., Negas, T. and Cook, L.P., 1981 *Phase Diagrams for Ceramists*, Vol 5, G. Smith, Ed, the American Ceramic Society, Inc, Columbus, Oh, Figure 5024.


<table>
<thead>
<tr>
<th>Table 1: Turkey Point Assembly B17 Fuel Characteristics*</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Cladding Material Burnup (GWD/MTu)</td>
</tr>
<tr>
<td>Turkey Pont #3 Zircalloy-4 ~27-Fuel Column Middle</td>
</tr>
<tr>
<td>Fuel Enrichment Original Density Cladding Thickness</td>
</tr>
<tr>
<td>2.559 wt% U\textsuperscript{235} 92% 0.0243 in.</td>
</tr>
<tr>
<td>Average Post-irradiation Grain Size (\textmu m) Average Rod Internal Pressure (psi) Fission Gas Release</td>
</tr>
<tr>
<td>~23 373 ± 27 0.3%</td>
</tr>
<tr>
<td>Average Core Linear Power (W/Cm) Years out of Core</td>
</tr>
<tr>
<td>182 7</td>
</tr>
</tbody>
</table>

*Davis and Pasupathi, 1981
TABLE 2
CHARACTERISTICS AND ROD SAMPLES

<table>
<thead>
<tr>
<th>Defect Size (mm)</th>
<th>Test Temperature (°C)</th>
<th>Relative Burnup</th>
<th>Diametral Strain During Defecting %</th>
<th>Location of Initial Defect From Bottom of Rod (mm)</th>
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<tr>
<td>760</td>
<td>250</td>
<td>1</td>
<td>&lt;0.2, 0.64, 0.28, 0.48, 0.56</td>
<td>1543, 1727, 2559, 2965, 2762</td>
</tr>
<tr>
<td>37</td>
<td>250</td>
<td>1</td>
<td>0.02, 0.07</td>
<td>1321, 1746</td>
</tr>
<tr>
<td>12</td>
<td>250</td>
<td>1</td>
<td>0.03, 0.08</td>
<td></td>
</tr>
<tr>
<td>760</td>
<td>283</td>
<td>0.3</td>
<td>0.03, 0.08</td>
<td></td>
</tr>
<tr>
<td>12</td>
<td>283</td>
<td>1</td>
<td>0.28, 0.03</td>
<td></td>
</tr>
<tr>
<td>26</td>
<td>283</td>
<td>0.8</td>
<td>0.72, 0.08</td>
<td></td>
</tr>
<tr>
<td>760</td>
<td>283</td>
<td>1</td>
<td>0.64, 0.07</td>
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<tr>
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<td>325</td>
<td>1</td>
<td>0.48, 0.05</td>
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<tr>
<td>8</td>
<td>325</td>
<td>1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>760</td>
<td>360</td>
<td>1</td>
<td>0.56, 0.06</td>
<td></td>
</tr>
<tr>
<td>27</td>
<td>360</td>
<td>1</td>
<td></td>
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(a) Based on gamma scans of companion rods
TABLE 3

INCUBATION TIME AND CRACK GROWTH RATE FOR DEFECTED ROD SEGMENTS

<table>
<thead>
<tr>
<th>Temperature (°C)</th>
<th>Defect Size (mm)</th>
<th>Location (inches from fuel column)</th>
<th>Incubation Time (hrs)</th>
<th>Crack Growth Rate (cm/min)</th>
</tr>
</thead>
<tbody>
<tr>
<td>360</td>
<td>760</td>
<td>108</td>
<td>20</td>
<td>2.3 x 10^{-3}(a)</td>
</tr>
<tr>
<td>360</td>
<td>27</td>
<td>84</td>
<td>52-60</td>
<td>4.2-8 x 10^{-3}</td>
</tr>
<tr>
<td>325</td>
<td>760</td>
<td>116</td>
<td>79</td>
<td>1.4 x 10^{-3}</td>
</tr>
<tr>
<td>325</td>
<td>8</td>
<td>76</td>
<td>455</td>
<td>1.4 x 10^{-3}</td>
</tr>
<tr>
<td>283</td>
<td>760</td>
<td>4</td>
<td>&lt;140, ~40</td>
<td>1 x 10^{-3}</td>
</tr>
<tr>
<td>283</td>
<td>12</td>
<td>100</td>
<td>1125</td>
<td>4.8 x 10^{-4}</td>
</tr>
<tr>
<td>283</td>
<td>26</td>
<td>124</td>
<td>~210</td>
<td>4 x 10^{-4}</td>
</tr>
<tr>
<td>283</td>
<td>760</td>
<td>68</td>
<td>830</td>
<td>3.0 x 10^{-4}</td>
</tr>
<tr>
<td>250</td>
<td>37</td>
<td>68</td>
<td>~5100</td>
<td>--</td>
</tr>
</tbody>
</table>

(a) Split took a radial turn after 50 hours.

(b) In both directions from the defect (total length increase rate).
<table>
<thead>
<tr>
<th>Test Temperature (°C)</th>
<th>Defect Size (mm)</th>
<th>Initial Diameter at Defect (in.)</th>
<th>Breach (%)</th>
<th>Type of Breach</th>
</tr>
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<tbody>
<tr>
<td>360</td>
<td>27</td>
<td>.4185</td>
<td>0.69</td>
<td>Flap</td>
</tr>
<tr>
<td>360</td>
<td>760</td>
<td>.4185</td>
<td>6.83</td>
<td>Axial</td>
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<tr>
<td>325</td>
<td>8</td>
<td>.4179</td>
<td>1.0</td>
<td>Spiral</td>
</tr>
<tr>
<td>325</td>
<td>760</td>
<td>.4183</td>
<td>7.49</td>
<td>Flap</td>
</tr>
<tr>
<td>283</td>
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(a) Split occurred at a location away from the defect. Strain at defect location was 0.29 percent.

(b) Split has widened too much to make true strain measurement. NM - not measured
FIGURE 1:
Typical Defects. When the cladding is pressurized, the defect location is indicated by the fine spray of kerosene placed on the cladding surface.
FIGURE 2:
Typical Sample of Bare Fuel Fragments Prior to Testing
FIGURE 3:
Weight Change of Bare Fuel Fragments Soaked in Air as a Function of Time-At Temperature
FIGURE 4:
Cure Fuel Fragments Before Testing and After 12.5 Hours at 300°C in an Air Atmosphere.
FIGURE 5:
Time to Bare Fragment Spallation
FIGURE 6:
Comparison of Time-to-0.6% Weight Gain for Irradiated CANDU and Turkey Point Fuel
FIGURE 7:
Determination of Cladding Splitting Velocity and "Splitting" Incubation Time from Crack Propagation Data.
FIGURE 8:
Cladding "Splitting" Incubation Period for Different Temperatures and Defect Sizes Compared to Bare Fragment "Spallation" Incubation
FIGURE 9:
Modes of Cladding Defect Propagation
FIGURE 10:
Velocity of Cladding Splitting or Oxidation Front Movement as a Function of Temperature
BEHAVIOUR IN AIR AT 175-400°C OF IRRADIATED UO\textsubscript{2} FUEL

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ABSTRACT

We have extended our study of irradiated, defected UO\textsubscript{2} fuel elements to 200 and 400°C. At 200°C there was no diametral change, but at 400°C we observed swelling and severe sheath splitting. Neither short-lived fission products, nor Cs-134, Cs-137 or Ru-106 above background, were detected. Maximum Kr-85 release was <7.4 \times 10^4 Bq (\leq 2 \times 10^{-6} Ci). Discharge time was 2.5 years.

We have extended our UO\textsubscript{2} fragment studies to 400°C. The oxidation process for unirradiated and irradiated fuel up to 300°C was characterized by activation energies of 140 \pm 10 and 120 \pm 10 kJ/mol, respectively; enhancement of oxidation rate was confirmed in the irradiated samples. There is an apparent reduction of activation energy above about 300°C.

Fuel elements with artificial and natural defects showed similar oxidation and dimensional response at 250°C. Behaviour of fuel fragments from the defect area of a naturally-defected element is consistent with that for fragments from intact elements when prior oxidation during the defect period is considered.
1. INTRODUCTION

The post-irradiation oxidation behaviour of $\text{UO}_2$ fuel in air up to about 400°C is attracting current attention because of its relevance to station fuel-handling, spent fuel transportation and dry storage. Noting a gap in literature data in this range, we previously reported air-oxidation results for defected irradiated $\text{UO}_2$ fuel elements in 220-250°C [Hastings and Novak, 1983a; Novak et al., 1983] and irradiated $\text{UO}_2$ fragments from 175-25°C [Hastings and Novak, 1983b]. Further, we showed the strong enhancement of oxidation in irradiated compared with unirradiated fuel [Novak and Hastings, 1983; Hastings and Novak, 1983b]. Other literature work had been in higher temperature ranges, usually above 300°C, or for forms of $\text{UO}_2$ not typical of fuel (e.g., fine powders), as outlined in an earlier review [Novak et al., 1983].

In this paper we extend our irradiated fragment and element studies to 400°C, and include data on unirradiated fragments. In addition to dimensional stability and fuel structure, we also examined fission product release at 200°C and 400°C, and oxidation behaviour of previously-defected fuel.

2. EXPERIMENTAL

2.1 FUEL EXAMINED

The fuel was CANDU-design $\text{UO}_2$ of nominal starting density 10.6 Mg/m$^3$. Irradiated Bruce, Pickering and Douglas Point NGS (Nuclear Generating Station) fuel was studied. Table 1 gives characteristics of the fuel. In summary, we examined:

1. Bruce and Pickering NGS fragments at 275-400°C, including fragments from previously defected fuel. Unirradiated fragments were also examined.
2. Bruce NGS elements at 200 and 400°C, dimensional stability and fission product release.

3. Bruce and Douglas Point NGS elements with artificial and natural defects, including oxidation behaviour of fragments from naturally-defected Douglas Point NGS fuel.

Figure 1 shows the position of defects deliberately introduced into elements before the oxidation study. Defect holes were 0.8 mm in diameter, with one or six holes per element.

2.2 HEATING

Irradiated full-sized elements and fragments were heated in-cell using a Lindberg three-zone tube furnace with a mild steel sleeve to minimize temperature variation. Unirradiated fragments were heated in a Sybron Thermolyne furnace. Unirradiated and irradiated fragments were placed in crucibles with a loose-fitting refractory lid to minimize particle dispersion. Pre-test commissioning and actual operation showed good temperature stability. Control was generally ± 5°C over the range of temperatures used (175-400°C) for times up to about 700 h. Cell air was typically at 30°C, with a relative humidity of 50-55%, laboratory air was 20-25°C, with a similar relative humidity.

2.3 WEIGHT MEASUREMENTS

Weight changes on unirradiated and irradiated fragments, were measured using a Mettler electronic balance of 400 g capacity, with reproducible readings to ± 1 mg. Weight changes in elements were measured in-cell using an Ohaus Harvard balance of 2000 g capacity, capable of reproducible readings to ± 10 mg. We experienced some problems with irradiation-induced drift with the in-cell balance, but were able to compensate by using standard weights for comparison.
2.4 DIMENSIONAL CHANGES

Diametral changes on individual elements were measured by micrometer at a minimum of seven different positions on selected elements, at 0°C and 90°C. Positions of measurement were at defect regions, and remote from the defects. A minimum of three readings was taken at each position and the result averaged.

2.5 METALLOGRAPHY

After visual examination, including photography, selected irradiated full-sized elements were sectioned for metallographic examination and determination of local 0/U values. Transverse sections were cut from the areas with one and four defects, and from areas remote from the defects. Sections were etched to reveal grain structure and degree of oxidation. Samples for determination of 0/U values were generally taken adjacent to the metallographic samples. The standard ASTM method was used for 0/U determination.

2.6 FISSION PRODUCTS - OXIDIZING SWEEP

Figure 2 shows a schematic diagram of the experimental system for oxidizing sweep studies, similar to that used in other testing [Lorenz et al., 1978; Lorenz et al., 1980]. All tests were carried out in the Special Project Hot Cell Facility, Chalk River Nuclear Laboratories. The technique depends on a carrier gas (air) flowing over the defected element at temperature, past a spectrometer, then to a series of traps. The fuel element was in a 2.5 cm inside diameter stainless steel tube enclosed in a mild steel sleeve within a Lindberg three-zone furnace. Control was generally ± 5°C for the test period. Chromel/alumel type K thermocouples were attached outside and inside the stainless steel tube to monitor the temperature. The stainless steel tube extended 15 cm on each end of the furnace so that the ends were cold. All other piping was polyethylene tubing which resists
picking up activity. Volatile fission products were monitored on-line in a loop of piping which extended out of the cell, into a lower radioactive background area. This optimized the sensitivity of the ND-66 gamma-spectrometry system which uses a Ge detector of high efficiency, ~13% for Co-60. The system is sensitive to \(<10^3\) Bq (\(<3 \times 10^{-8}\) Ci). Following the spectrometer were a series of filters consisting of particulate, CdI\(_2\), iodophenol, silver xerolite and charcoal, all at cell ambient temperature to trap iodine. The last trap in the system was charcoal at \(-70^\circ\)C to trap all Xe, Kr, and any iodine which escaped the first series of traps. No fission products can escape through these traps. After an oxidation run, all pipework from the furnace tube to the cold trap was counted for deposited fission products.

Flowmeters at the start and end of the flowline checked that no leaks developed during the experiment. Fuel elements were brought to temperature under argon. The argon in the furnace tube was then rapidly displaced by air flowing at 1 L/min for 5 min and the flowrate was reduced to \(\approx300\) mL/min to optimize the sensitivity of the gamma spectrometry. The flow was selected to ensure particulate matter was not transported in the system. A factor of at least two higher in flow rate is required to move particulates. Sufficient air was supplied at all times to ensure oxygen starvation did not occur.

3. RESULTS & DISCUSSION

3.1 DEFECTED, IRRADIATED ELEMENTS AT 200-400°C

We previously showed [Novak and Hastings, 1983; Novak et al., 1983] irradiated CANDU fuel elements with single and multi-defects maintained reasonable dimensional stability up to 685 h at 220 and 230°C. This is consistent with oxidation primarily up to U\(_3\)O\(_7\). Elements with single and multiple defects showed significant diametral increases (2-12%) and severe sheath splitting after about 200 h at 250°C, consistent with oxidation primarily to U\(_3\)O\(_8\).
3.1.1 Bruce NGS Element 12 Bundle G00063 at 200°C for 24 h

Element weight prior to the test was 640 g. Post-test weighing showed no change. Similarly, there were no dimensional changes; pre- and post-test diameter was 13.02 ± 0.01 mm. No short-lived fission products were detected at the spectrometer during the test, or on traps or piping after the test. Table II gives results for longer-lived species. Kr-85, Cs-134, Cs-137 and Ru-106 resulting from test operation were looked for, but only an insignificant amount of Kr-85, less than 5.9 x 10^4 Bq, was detected above background. We had constant Cs-134, Cs-137 and Ru-106 counts during the test, originating from the cell background.

The dimensional and fission product stability after 24 h at 200°C is not surprising. Previous element tests at 220°C [Hastings and Novak, 1983a; Novak et al., 1983] and UO2 fragment tests at 200°C [Hastings and Novak, 1983b] confirm no significant changes after only 24 h. In element tests at 220°C, more than 500 h was required to produce a weight increase of 0.5% [Novak and Hastings, 1983; Novak et al., 1983]. In fragment tests at 200°C, the weight increase was only 0.02% after about 160 h [Hastings and Novak, 1983b]. Significant oxidation to U3O8 is not expected until about 250°C. Also, at 200°C, gamma-active fission product transport is not expected. Diffusion would be extremely low at this temperature. The slow rate of oxidation would also minimize any fission product transport by mechanisms such as grain boundary separation and increased oxygen content of the UO2 lattice. No short-lived species would be measurable after such a long discharge period.

3.1.2 Bruce NGS Element Bundle G00063 at 400°C for 24 h

During the test at 400°C, no short-lived fission products were detected at the spectrometer, or on traps or piping after the test. Table III gives results for longer-lived species. Krypton-85, Cs-134, Cs-137 and Ru-106 resulting from test operation were looked for, but only 7.4 x 10^4 Bq of Kr-85 was detected, slightly more than at 200°C.
We only observed cell background activity of Cs-134, Cs-137 and Ru-106 during the test. Additionally, there was no evidence for transport of particulate oxidized fuel out of the element into the sweep system, despite the high degree of fuel oxidation observed. The flow rate was carefully selected to minimize particulate transport. Though at 400°C gaseous diffusion is still low, the degree of oxidation to U₃O₈ could have exposed sufficient fresh surfaces to allow release. The short-lived species have decayed to non-measurable values since discharge, but one might expect some small release of longer-lived gases during the oxidation conditions of the test. However, the insignificant amount of Kr-85 detected suggests that transport of these species is also limited. There would be even less tendency to transport under stagnant air conditions.

Post-test visual examination revealed severe sheath splitting at both the ends (one defect hole) and mid-plane (four-defect holes), as shown in Figure 3, consistent with local oxidation of UO₂ to U₃O₈. Figure 4 shows detail of the major split at mid-plane. Diametral increases ranged from 4-8% at the ends (single defect) to in excess of 50% at the major split at the mid-plane. The diametral increases were localized in the vicinity of the deliberate defects; intermediate between the mid-plane and ends, the diameter was very close to the pre-test value. This localized oxidation has been observed in other element tests [Hastings and Novak, 1983a; Novak et al., 1983]. The results indicate that, in handling defected fuel with a 2.5 year discharge time, problems from gaseous fission product release should not arise in periods up to 24 h at 400°C. However, sheath deterioration, and possible movement of particulate oxidized material, will have to be considered under conditions appropriate to fuel handling, transport and storage.

In Figure 5, we summarize data on defected element dimensional stability in air at 200-400°C, for times up to 685 h. In the case of elements with six defects, metallography has shown that oxidation proceeds independently at or near the one-defect regions at the element ends, and at the four-defect regions [Novak and Hastings, 1983; Novak et al., 1983]. Results from irradiated elements with one defect are also plotted. Clearly, there is a significant change between multi-
defect case at 250°C. The comparatively stable element dimensions below 250°C are explained by the formation of the non-swelling U$_3$O$_7$ phase. The volume change that accompanies conversion to U$_3$O$_8$ after about 200 h at 250°C is sufficient to strain the sheath significantly to 5-12%, accompanied by severe splitting. No severe splitting was observed during intermediate weighings at 118 h; the split thus occurred between 118 and 208 h. Note that in unirradiated, stainless-steel-clad, AGR fuel with one deliberate defect, severe splitting occurred after 600 h at 250°C [Ainscough and Oldfield, 1964]. However, within 24 h at 400°C, significant sheath splitting and diametral increase has occurred in our case.

From the data at 220 and 230°C, no dimensional increase exceeds about 1%. At 250°C, one-hole defects from Pickering produce diametral increases of about 2%. If we accept this as the limit above which significant sheath splitting begins to occur, then defected, irradiated elements will remain dimensionally stable (i.e. diametral increases less than 2%) for times up to at least 700 h at 200°C, 500 h at 230°C, between 120 and 200 h at 250°C, in which interval severe splitting occurred, and less than 24 h at 400°C.

3.2 IRRADIATED AND UNIRRADIATED FRAGMENTS AT 175-400°C

We previously reported oxidation data for Pickering and Bruce NGS irradiated fragments from 175-250°C [Hastings and Novak, 1983b]. Figure 6 shows data from the present study on Pickering NGS irradiated fragments from Bundle H22228, at 300-400°C, and Pickering/Bruce NGS data at 275°C, with ΔW (weight increase as a percentage of original sample weight) as a function of heating time in air. Figure 7 shows data for unirradiated Pickering NGS-design UO$_2$ fuel fragments in the range 275-400°C. Previously [Hastings and Novak, 1983b] we gave results at 230 and 250°C. At 175°C in the current study, ΔW after 775 h was 0.03%, rate of weight changes F, = 0.4 x 10^{-4} %/h. Fragment size was uniform at about 1 g to minimize the known effects of variations in surface-to-volume ratio. Total sample weight was typically 35 g; calculated pre-test surface area was about 50 cm
We have found weight increase (as a percentage of starting weight) to be a useful defining parameter. Other studies [Simpson and Wood, 1983] have used mg/cm$^2$ (surface area)/time, but this is complicated by the rapid changes in fragment surface area as oxidation occurs.

Figure 6 shows an increasing rate of weight gain with temperature in the range 275-400°C for irradiated fragments; Figure 7 shows a similar trend of weight increase with temperature, for unirradiated fragments, but at a rate about a factor of three lower for a given temperature and time. We noted previously [Novak and Hastings, 1983; Hastings and Novak, 1983b] the irradiation-enhancement of oxidation in both fragments and elements. At 400°C, for unirradiated fragments, there was 75% conversion to U$_3$O$_8$ in about 3 h; for irradiated fuel a similar conversion is achieved in about 1 h. We assume a 1% weight increase corresponds with about 25% conversion to U$_3$O$_8$. One expects a reduction in the rate of conversion as the 100% level is approached. Previous data [Novak and Hastings, 1983; Novak et al., 1983] suggested that, for the timescale of our study, the primary oxidation phase below 230°C was U$_3$O$_7$; above 250°C, the primary phase was U$_3$O$_8$. In recent preliminary work [8] on fragments from irradiated light water reactor (LWR) fuel, directly comparable with our own study, weight gain was 1.4-2.2% after 350 h at 230°C; we observed 2.0-2.5% for the same conditions. Note that we did not observe any large incubation period (>50 h) for unirradiated or irradiated fragments, in contrast with results reported for unirradiated fuel [Simpson and Wood, 1983].

Figure 8 shows Arrhenius plots of F, the rate of weight increase (%/h) (175-400°C), as a function of reciprocal absolute temperature (T) for the unirradiated and irradiated fuel fragment data from Figures 6 and 7, and previous studies [Hastings and Novak, 1983b]. F was obtained assuming a linear weight change at each temperature. The fit up to 300°C was given by:

$$F = 1 \times 10^{12} \exp\left(-\frac{140 \pm 10}{RT}\right) \text{ %/h}$$
for unirradiated fuel, and

\[ F = 2.23 \times 10^{10} \exp - \frac{120 \pm 10 \text{kJ/mol}}{RT} \%/h \]

for irradiated fuel, with correlation coefficients of 98% and 99%, respectively, \( R = 8.3 \text{ J/mol.K} \). Previously [Hastings and Novak, 1983b] we had reported \( 130 \pm 10 \text{kJ/mol} \) from 175-250°C, for irradiated fragments. The results also suggest a change in activation energy above about 300°C for both unirradiated and irradiated fragments, but data are limited to three points. X-ray studies [Taylor et al., 1980] on surface oxidation of unirradiated UO₂ at 225-275°C in air obtained an activation energy of 120 kJ/mol for the process. Recent oxidation data [Simpson and Wood, 1983] on unirradiated UO₂ fragments suggested an activation energy of \( 160 \pm 5 \text{kJ/mol} \) for temperatures less than 320°C and \( 100 \pm 7 \text{kJ/mol} \) at 320 - 450°C, apparently reflecting a change in activation energy proposed earlier from defected element work [Boase and Vandergraff, 1977]. The change has been attributed to a change in stability of beta-U₃O₇ (\( \gamma_1 \)) above about 350°C [Simpson and Wood, 1983]. However, we have evidence that the turnover at 300°C may be due to experimental technique. Preliminary tests at 350°C show the rate of weight change for unirradiated and irradiated fragments in a lidless crucible is about five times that for fragments in a crucible with a lid. This implies oxygen starvation at the high oxidation rates above 300°C, in the latter cases. Confirmatory studies are being performed. The effect of irradiation on enhancing oxidation is clearly shown in Figures 6-8.

In a previous "worst case" analysis [Hastings and Novak, 1983b] of fragment oxidation, using conservative assumptions, we suggested 15% conversion to U₃O₈ was sufficient to produce a 2% diametral increase in a defected fuel element, below which there was no evidence of sheath splitting. If we assume our relationship of \( F, T \) and \( Q \) (activation energy) extends to 150°C, a typical storage temperature, then the 15% conversion criteria would be reached in about four years, though the time is sensitive to \( Q \). In long-term canister storage studies, involving defected fuel, while there is evidence of
some localized oxidation of the fuel surface in the defect after 2.5 years at 150°C [Wasywich et al., 1984], there is no sign of diametral change or sheath splitting. Our fragment studies clearly give a conservative result. In practice, there is evidence from higher temperatures [Novak and Hastings, 1983; Novak et al., 1983] that oxidation in elements is localized within about one pellet length of the defect; oxide formation apparently blocks access of air to the rest of the fuel.

Any direct comparison of oxidation results from CANDU and LWR fuel with significantly different power histories (and thus microstructures) should be made with caution, as should any conclusions based on unirradiated fuel tests alone, as outlined elsewhere [Novak and Hastings, 1983; Hastings and Novak, 1983b].

3.3 BEHAVIOUR OF NATURALLY-DEFECTED IRRADIATED FUEL

3.3.1 Element Behaviour

Most reported testing has been carried out on irradiated fuel elements with artificially-produced (usually drilled) defects. Figure 9a shows sheath splitting in Bruce NGS element, bundle F13566, with a single artificial defect, after 215 h at 250°C in air. The split is about 3 cm long, extending about 1.5 cm on either side of the original hole. The maximum diametral increase is about 8%. Figure 9b shows a metallographic cross-section at the split, 2 mm from the original defect. Oxidation has occurred preferentially at grain boundaries and fuel cracks, extending radially not only from the defect but also from circumferential positions remote from the defect. As we polished closer to the original defect, we experienced almost complete "pull-out" of fuel across the section, consistent with oxidation at grain boundaries and cracks. However, metallographic sections 5 cm from the end of the crack, and 5 cm from either end of the element did not show evidence of severe oxidation. O/U at the split was about 2.3 ± 0.1; O/U at the end positions was 2.0. Fuel typically has a
starting 0/U of 2.005 ± 0.005. We would expect a higher 0/U for complete conversion to \( \text{U}_3\text{O}_8 \), but potentially high 0/U material was lost from the vicinity of the split. There was about 2 mm element bowing at the mid-plane.

Figure 10 shows the evolution of sheath splitting in a naturally-defected Douglas Point NGS element, bundle G01507, at 250°C: (a) as-irradiated, (b) 155 h, (c) 173.5 h, and (d) 194 h. There was no splitting visible at 43.5 h; first evidence of fine splits was observed at 110.5 h. There were no significant diametral increases up to, and including, 155 h. At 173.5 h, the maximum diametral increase in the vicinity of the defect hole was about 2%. At 194 h it was in excess of 10%; the exact value was difficult to determine because of the split configuration. The diametral increase was localized; less than 2 cm from either end of the sheath split, measured diameters were identical with pre-test values. Measurements midway between the element mid-plane and ends, and at the element ends, showed no changes. Centreline bow was about 8 mm. 0/U measured at the crack was 2.1 ± 0.1; values at both element ends were 2.01. The 0/U at the crack is unexpectedly low, as \( \text{U}_3\text{O}_8 \) has clearly produced the significant swelling. However, as with the Bruce NGS element, the measurement is a "smeared" one, and a significant amount of material of potentially high 0/U from the vicinity of the split was lost in preparation. Figure 11, a metallographic section through the crack, at the defect position, shows how the oxidation has proceeded radially, similar to that shown in Figure 8, and observed previously in artificially defected fuel [Novak and Hastings, 1983; Novak et al., 1983]. Metallography also confirmed that significant oxidation was localized at the split. Oxidation elsewhere in the fuel, including at the primary defect at the end cap was localized, at the surface as shown in Figure 12. Prior experience suggests that the defect was a "pin-hole" that quickly re-sealed.

Behaviour of the naturally-defected element was closely similar to that of the Bruce NGS element with a single artificial defect at 250°C, and also that of a multi-defect element at 250°C [Novak and Hastings, 1983; Novak et al., 1983], in which severe sheath
cracking occurred after 208 h at 250°C at four defect holes at the element mid-plane. In the latter two cases, powers were significantly higher (up to 45 kW/m) than that for the Douglas Point NGS element (30 kW/m). Measured diametral increases in the vicinity of the split were similar in all three cases. The result is an indication that results from artificially defected elements can be applied to naturally-defected elements.

3.3.2 Fragment Behaviour

Preliminary tests have been carried out at 230°C on fragments extracted from the region of a natural defect in a Douglas Point NGS element. Fuel element characteristics were identical with those shown in Table I, except burnup was about 500 MW.h/kgU at a linear power range of 30-50 kW/m. Figure 13 shows the weight increases for fragments from the defect area and from an area more than 20 cm away, up to about 500 h; data on control fragments from an intact Pickering NGS irradiated fuel element at 230°C are included for comparison. Weight change and rate of weight change in the fragments from the defect area of naturally defected elements were about a factor of two less, compared with the fragments remote from the defect or from intact fuel. In the former case, this difference may reflect the prior oxidation of the fuel in the reactor channel while defected. Time in-reactor following the defect could also be a factor.

4. CONCLUSIONS

1. At 200°C in air, there was no diametral or weight change in a defected, irradiated element; at 400°C there was severe sheath splitting and diametral increase in less than 24 h. Results correlate well with previous tests.

2. At 200 and 400°C neither short-lived fission products nor Cs-134, -137 or Ru-106 above background, were detected in fuel
with 2.5 y discharge times. Maximum Kr-85 release was $\leq 7.4 \times 10^4$ Bq ($\leq 2 \times 10^{-6}$ Ci) at 400°C.

3. In fragment studies at 175-400°C, the oxidation process for unirradiated and irradiated fuel up to 300°C was characterized by activation energies of 140 + 10 and 120 + 10 kJ/mol, respectively; enhancement of oxidation rate in irradiated samples was confirmed.

4. There is an apparent reduction in activation energy above about 300°C, but this may be related to experimental technique. Confirmatory studies are in progress.

5. No large incubation period was observed for either unirradiated or irradiated fuel in the range 175-400°C.

6. Fuel elements with artificial and natural defects showed similar oxidation and dimensional response at 250°C in air.

7. Preliminary results indicate that behaviour of fragments from the defect area of a naturally-defected element is consistent with that for fragments from intact elements, when prior oxidation during the defect period is considered.

5. ACKNOWLEDGEMENTS

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6. REFERENCES


### TABLE I: CHARACTERISTICS OF FUEL EXAMINED

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**OPERATING CONDITIONS**

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<td>1978 Nov. 10</td>
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* *Nuclear Generating Station*  
**Cladding is collapsible during irradiation; post-irradiation fuel-to-sheath gap is about 0.1 mm.**  
***Outer element values; inner element values 17% less.**  
+Douglas Point NGS fuel and operating conditions are similar to those of Pickering NGS. Douglas Point NGS element in current test operated at 28-32 kW/m to 160 MW.h/kgU. Discharge date was 1981 June 25; Bundle G10507
TABLE II: INTEGRATED RELEASE, BRUCE NGS ELEMENT 12 BUNDLE G00063, 24 h AT 200°C.

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<th>Fractional Release</th>
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<td>4.2</td>
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<td>61.7</td>
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<td>Cs-137**</td>
<td>52.8</td>
<td>14.3</td>
<td>&lt;3.7</td>
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* Calculated from power history, based on 585 g UO2
** Cell background levels - no contribution during test

TABLE III: INTEGRATED RELEASE, BRUCE NGS ELEMENT 7 BUNDLE G00063, 24 h AT 400°C.

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<td>≤7.4</td>
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<td>61.6</td>
<td>16.6</td>
<td>&lt;111.0</td>
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<tr>
<td>Cs-134**</td>
<td>15.8</td>
<td>4.3</td>
<td>&lt;111.0</td>
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<tr>
<td>Cs-137**</td>
<td>55.4</td>
<td>15.</td>
<td>&lt;11.1</td>
</tr>
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* Calculated from power history based on 585 g UO2
** Cell background levels - no contribution during test
FIGURE 1:
Position of artificial defects. Six-defect case is shown. One-defect case has hole at element mid-plane. Not to scale.
FIGURE 2:
Schematic diagram of sweep system.
FIGURE 3:
Severe sheath splitting after 24 h at 400°C in air; defect areas arrowed. Element length is 500 mm.
FIGURE 4:
Detail of mid-plane split in Figure 3, 24 h at 400°C in air.
FIGURE 5:
Diametral increase as a function of time, defected elements at 200-400°C in air.
FIGURE 6:
\( \Delta W \) (weight increase) of irradiated UO\(_2\) fragments as a function of heating time, 275-400\(^\circ\)C in air. We previously reported data at 175-250\(^\circ\)C [Hastings and Novak, 1983b].

\[
\begin{align*}
\Delta W &\ (weight\ increase)\ of\ unirradiated\ UO_2\ fragments\ as\ a\ function\ of \\
&\text{heating time, 275-400\(^\circ\)C. We previously reported data at } \\
&175-250\(^\circ\)C. \Delta W\ after\ 775\ h\ at\ 175\(^\circ\)C\ was\ 0.03\%\ for\ F = 0.4 \times 10^{-4}\ \%/h. 
\end{align*}
\]
FIGURE 8:
Rate of weight increase as a function of reciprocal absolute heating temperature, unirradiated and irradiated fragments, 175-400°C in air.
FIGURE 9:
(a) Sheath splitting in irradiated Bruce NGS element with artificial defect, 215 h at 250°C in air, (b) metallographic cross-section through split.
FIGURE 10:
Evolution of sheath splitting in irradiated naturally defected Douglas Point NGS element at 250°C: (a) as-irradiated, (b) 155 h, (c) 173.5 h and (d) 194 h.
FIGURE 11:
Metallographic cross-section through split in Figure 10(d).

FIGURE 12:
Localized surface oxidation remote from defect, Douglas Point NGS element.
FIGURE 13:
Weight increases for fragments from the defect area and from an area 20 cm from the defect, at 230°C in air. Data from control fragments are also shown.
SESSION 7

PANEL DISCUSSION: DRY STORAGE - THE FUTURE

Chairman: R.W. Lambert

Electric Power Research Institute
Panelists

A.B. Johnson, Jr., Pacific Northwest Laboratory (United States)
D. Deacon, GEC Energy Systems Limited (United Kingdom)
P. Le Sueur, Electricité de France (France)
R.E. Einziger, Westinghouse Hanford Company (United States)
S.J. Naqvi, Ontario Hydro (Canada)
A. Future Dry Storage R&D

Dry storage of spent fuel has several scenarios: temporary or lag storage; interim storage; monitored retrievable storage; disposal of spent fuel also has direct relevance to dry storage. My comments are directed principally to interim storage, but also largely apply to the other storage regimes.

Complete the data base for dry storage of Zircaloy-clad fuel in inert gas; tests and demonstrations now underway or completed seem to provide sufficient basis for LWR fuel, including work at E-MAD (Nevada) and Battelle-Columbus in the USA and at several locations in the Federal Republic of Germany.

Finalize thermal and radiation code assessments for dry storage.

Complete the data base on behavior of cladding defects and water-logged fuel in dry storage. Tests are underway in Canada, the FRG, and the USA. Inclusion of fuel with defective cladding in inert gas dry storage demonstrations should be considered.

Identify a maximum temperature for storage of LWR fuel in air. Work is underway in Canada, the UK, and the USA. Predictions of an acceptable temperature have decreased. Relatively long times may be required to demonstrate a licensable air storage temperature.

Demonstrate dry storage of consolidated fuel.

* No written contribution to the Panel Discussion was provided by R.E. Einziger (Westinghouse Hanford).
Develop advances in dry storage monitoring and/or established confidence that minimizes the need for monitoring.

Consider areas of storage system (e.g., cask, silo, etc.) performance which need R&D, and pursue areas which need to be addressed. An example for possible assessment is the question of storage/shipping cask recertification at the end of extended storage. What, if any, recertification will be required prior to shipment? If a need for recertification is likely, what actions will facilitate the process (e.g., NDT, specimen tests, etc.).

B. Where are the Greatest Opportunities?

An issue not yet addressed is demonstration of dry storage of consolidated fuel, which has potential application in all storage regimes.

Definition of acceptable air storage conditions is an area of continuing interest, perhaps for storage applications, but also to define a storage temperature below which monitoring for inertness could be relaxed for fuel storage in inert cover gases.

C. What are the Greatest Impediments?

For rod consolidation, there is insufficient "hard" information regarding process costs and waste management aspects for hardware disposal.

Recertification of a storage cask for shipment following extended dry storage is difficult to assess because regulations a decade or more from now are difficult to anticipate.
A lot has been said at this conference about the technical merits, comparison of costs, licensing requirements, etc., of the various storage options.

A further factor that has had very little mention but I believe will have significant effect on the choice and location of the irradiated fuel store is the one of public opinion.

To find a site for an AFR in any country is likely to prove difficult.

If a new site is chosen then we are all familiar with the public cries of "Nuclear Dustbins" and "NIMBY" (not in my own backyard). If we state that the store is merely an intermediate store, then the question of increased fuel movements over the public highways has to be addressed.

It would seem to me that the compromise is to locate the stores at the reactor sites. These sites by definition are already licensed and equipped for the short-term storage of irradiated fuel. Modification of the operating licence would be needed but this would be a modification to an existing licence and not a new licence requirement.

To locate the intermediate/long-term store at the reactor site allows fuel movement between the reactor building and the store to take place within the licensed site and therefore allows more flexibility.

After irradiated fuel has been stored for some decades in modular vault dry stores (MVDS), I believe that public confidence will have increased such that applications to increase the storage period will meet less opposition.
An MVDS for the life-time's fuel for a 1200 MW station is a very small construction, as has been shown in our presentation (Engineering and Safety Features of Modular Vault Dry Storage" in Session 3).

With respect to the technical merits of the various options for intermediate/long-term storage of the systems available, listening to the papers presented I would only comment that I believe preference should be given to those systems:

1. Which allow ready retrieval of the fuel.

2. Which allow ready monitoring of the state of the fuel and its containment envelope.

3. Which provide generous and reliable cooling such that fuel temperatures are low enough to allow air to be used as the containment environment.

4. Which generate the minimum volume of contaminated wastes in the process.

5. Where the technology and safety features have been demonstrated and licensed on similar plant(s).
As mentioned in my paper ("The French Utility's View on Irradiated Fuel Management" in Session 1) and in spite that it is not part of our basic principles, we know that long-term storage will be a necessity for the irradiated fuel assemblies for which reprocessing will be too expensive. Nevertheless, we hope that the quantity involved will not be very large.

In order to be ready when the time comes, Electricité de France is engaging studies for this long-term storage.

As far as a dry storage system is concerned, we can only give the main axes of the study which will be performed. Depending to the quantity to be stored and the forecast of the storage time, two ways will be studied.

First, the at-reactor storage can be envisaged in order to increase the storage capacity of the site. In this case, we will study steel, cast-iron or concrete containers. The fuel assemblies could be either unconsolidated or consolidated and the inside atmosphere will be air or inert gas. We think that it will be better for us to choose a container which could be licensed for both transport and storage.

Then away-from-reactor storage can also be envisaged for the irradiated fuel assemblies which will not be reprocessed. In this case, three solutions will be studied:

First the use of the previous containers placed in a large hall either at surface or underground; this solution has the advantage to limit the fuel handling.
Second, the use of canisters containing one or more fuel assemblies, consolidated or not, stored in dry wells drilled in the ground. In France, this solution has the big advantage to be well known because it has been used for a long time by the Commissariat a l'Energie Atomique for storing the glass blocks after the vitrification of the high activity reprocessing wastes.

Third, the use of the same canisters but placed into silos or vaults.

All these solutions will be carefully studied, taking into account the experience accumulated in the world and the technical means that we will have at our disposal at the time of the decision.

But the future begins tomorrow and we intend to get full scale dry storage experience for a small quantity of material. As a matter of fact, the 70 MWe gas cooled heavy water moderated reactor of Les Monts d'Arrée in Brittany will be definitely shut down next year and the decision is taken to store the core in dry conditions. The quantity involved is about 5,000 bundles, that is to say 50 t of uranium, which will be evacuated from the site in 1988 and 1989. Electricité de France and the Commissariat a l'Energie Atomique are studying the final dry storage solution, in particular in cooperation with the engineers of the Atomic Energy of Canada Limited, and this solution will be chosen by the end of this year. Then, the analysis of this experience will be very helpful for Electricité de France when the moment comes to make its choice for the dry storage of its PWR irradiated assemblies.
Although dry handling and storage of irradiated fuel has been practised under specific circumstances since the 1960's, and in that sense it's not a new technology, it is only recently that this technology has acquired prominence for providing complementary storage capacity for commercial reactor fuels. One of the main objectives of this workshop was to bring this fact in full light and Dr. Bert Johnson's review paper ("Dry Storage - Past, Present and Future" in Session 3) has done that admirably. Another fact of dry storage technology that needs better grasp and realization is that it has a wide range of application, e.g. from storage of freshly discharged fuel with appropriate cover gas and forced cooling, to the storage of older fuel in a totally passive system based on an air cover gas.

Further, the two main requirements of this technology i.e. proper cooling and containment of irradiated fuel are no different than that in most nuclear operations familiar in nuclear industry. Likewise the other requirements of fuel storage:

1. Maintenance of fuel integrity under variable environmental conditions;
2. Retrievability of irradiated fuel at all times;
3. Long-term durability of structures;
4. Operative care and surveillance;

are also not unique to dry storage and can be met adequately and in some cases even in a better way than in wet storage by following sound engineering design construction and operating principles.
In a recent invited paper at the ANS 1982 winter meeting, I made the point that no single dry storage method would be appropriate for every situation; it would rather vary with:

- fuel type
- irradiation history
- initial cooling
- defected and non-defected state of the fuel
- storage medium
- storage temperature
- mechanical stresses

This represents a wide range of situations and offers an interesting challenge to a designer.

Also while long-term irradiated fuel integrity is the major requirement here, it should be kept in mind that more than 99.9 percent of irradiated fuel (I/F) discharged from our reactors is defect free and has proven to maintain its integrity for long periods of storage under fairly variable environmental conditions. Thus any insistence on defining a single threshold storage temperature based on the result of defected fuel monitoring experiments would be an unjustified constraint on this technology.

It is possible by design to select the temperature at which one can store the fuel, which is one of the key parameters that was discussed at great length at this workshop. One can also select whichever storage medium is considered most appropriate: helium, argon, CO₂, or if the temperature is low enough, air. Further it is possible, in fact, to start off with an inert atmosphere and later switch over to air as the decay heat of I/F decreases with time.

I think the best approach with dry storage would be to follow a systems approach where we can combine the results of fuel integrity experiments with storage design principles and go on to demonstrate the
technology for licensing purposes. This is the approach we are following at Ontario Hydro rather than worry too much about one failure mechanism or be limited to the consideration of one dry storage concept.
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