CHAPTER 19

Storage and Disposal of Irradiated Fuel

prepared by

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Summary

Nuclear wastes, in particular irradiated nuclear fuel, must be handled, stored, and placed into permanent disposal facilities safely to prevent harm to people and the environment. Radioactive wastes can be grouped into four classes: (i) low-level radioactive waste, (ii) intermediate-level radioactive waste, (iii) high-level radioactive waste such as irradiated nuclear fuel, and (iv) uranium mine and mill waste. Storage and disposal of irradiated fuel in Canada follows a three-step process: storage in water-cooled pools, storage in air-cooled storage cylinders, and final disposal. The two stages of storage (the first two steps above) are fully proven and have been in practical operation for some time. The associated technical challenges that must be addressed and the engineered solutions are discussed in this chapter. Several configurations for final disposal have been shown to be technically feasible. Public acceptance and implementation remain to be achieved; ongoing work aimed at achieving these goals is discussed. Nature's "reactors" that existed billions of years ago are examined for useful analogies that can be applied to engineered disposal facilities.

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1 Introduction

This chapter focuses mainly on the challenges that must be met during storage and disposal of high-level nuclear waste, specifically storage and disposal of irradiated fuel (also called “spent” fuel or “used” fuel). Some illustrative potential solutions are also summarized.

To this end, Sections 2 to 5 describe the types of wastes, the overall strategy to manage them, the associated design considerations, and the potential mechanisms for damage to fuel integrity. Then each major phase of managing irradiated fuel is described: initial wet storage at reactor sites (Section 6), followed by dry storage at reactor sites (Section 7) and final disposal (Section 8). IAEA [2003] provides an extensive glossary of relevant terms.

This chapter draws heavily from previous publications. Significant passages that have been copied verbatim from other sources are enclosed within quotes “...”.

1.1 Learning Outcomes

The goal of this chapter is for the student to acquire a broad initial understanding of the main challenges that must be considered in designing facilities to store irradiated fuel (also called "spent" fuel), the evaluations performed, and the safety and other criteria used for selecting the principal features of the facilities for initial wet storage of irradiated fuel at reactor sites, subsequent dry storage of irradiated fuel at reactor sites, and the current Canadian concepts for eventual disposal of irradiated fuel (e.g., geological isolation).

2 Types of Waste

All biological life forms produce waste as part of their normal cycle of existence. Within the biosphere, dissimilar life forms (e.g., plants and animals) have evolved to exist in harmony and dependence on exchange of wastes. All human activity, starting with the most primitive prehistoric activity associated with subsistence, up to and including the most sophisticated industrial activity of today, produces waste. Regulations for managing all aspects of human-generated wastes have evolved from “non-existent” for most of human existence to the comprehensive and sophisticated regulations of today. Human industrial activity has not been in existence sufficiently long to test the ability of ecological systems to create synergies, interdependency, or both. Indeed, some industrial wastes are known to be detrimental to people and the environment. Such wastes must not be left untreated or allowed to enter the biosphere. Some nuclear wastes also fall into this category.

Under the terms of its statute, the International Atomic Energy Agency (IAEA) has a mandate to establish or adopt safety standards to protect people and the environment from the harmful effects of ionizing radiation. IAEA identifies six classes of nuclear waste ranked in order of their harmful radiological effects (from the lowest to the highest) and the escalating requirements for their safe disposal, as follows [IAEA, 2009]:

- Exempt waste
- Very short-lived waste

- Very low-level waste
- Low-level waste
- Intermediate-level waste
- High-level waste.

The crafters of the IAEA standard provided this list of waste classes as an aid to individual users, recognizing that it may not be necessary or suitable for all users to adopt the list explicitly. Users have the option of adopting the list literally or adapting it to their own needs, particularly users who have had prior experience with managing radioactive waste.

The Canadian Standards Association (CSA), in collaboration with industry, government, and the Canadian Nuclear Safety Commission (CNSC), has developed the "Radioactive Waste" standard [CNSC, 2015a] that recognizes four main classes of radioactive waste:

- Low-level radioactive waste
- Intermediate-level radioactive waste
- High-level radioactive waste
- Uranium mine and mill waste.

Although these two lists of radioactive waste classes are not identical, it can be readily discerned that the Canadian standard is not inconsistent with that of the IAEA. The Canadian standard omits "exempt" waste and combines the three classes of "low-level" waste into a single class. The two standards are explicitly consistent with respect to "intermediate" and "high-level" waste. The Canadian standard adds a separate class for "uranium mine and mill" waste, which reflects the importance of uranium mining in Canada. Every existing source of radioactive waste in Canada and its class is listed, and the entity responsible for its management is identified. Some entities may be responsible for managing more than one class of waste.

3 Strategies to Manage Radioactive Waste

<u>Low-Level and Intermediate-Level Wastes</u>

Radioactive waste has been produced in Canada since the early 1930s; see, for example, "Historic Nuclear Waste" [CNSC 2015a], which describes the period when radium and uranium were mined in the Northwest Territories and transported to Port Hope, Ontario, for refining. Subsequently, uranium in much greater quantities has been mined and milled, most notably in northern Ontario and Saskatchewan, and transported to Port Hope for refining and subsequent use in nuclear fuel.

The following steps are taken to manage low-level and intermediate-level radioactive wastes [CNSC, 2015b]:

- Decontamination and clean-up;
- Long-term storage and management by the user;
- Return to the manufacturer for long-term storage and management;

- Storage and management at large central facilities such as those operated or proposed by Ontario Power Generation and Canadian Nuclear Laboratories.

The remainder of this chapter deals with the management of irradiated CANDU fuel.

Irradiated Fuel

Irradiated fuel continues to produce power and emit radiation after it is removed from the reactor. The power and radiation decay with time, and therefore the need to cool the fuel and to monitor its activity also decrease with time. For this reason, irradiated nuclear fuel passes through two phases of storage before final disposal in the third phase, as follows:

- Immediately after its discharge from reactors, the fuel is stored for several years in deep cooling pools adjacent to the reactor.
- After several years of forced-circulation cooling in water, the fuel is transferred to concrete containers which are air-cooled by natural convection. The fuel can reside in these storage cylinders for up to 100 years before its ultimate disposal.
- In the third phase, irradiated fuel is intended to be permanently sequestered in specially designed facilities. As an illustrative example, one concept is to put the fuel in a sealed container which is surrounded by clay and placed in a room that is built deep underground in impervious rock.

Examples of these facilities, particularly as they apply to CANDU irradiated fuel, are described later in this chapter.

Irradiated CANDU fuel may well be commonly considered “waste” to be (eventually) permanently sequestered in specially designed facilities. At the same time, it also has the potential to be recycled to generate significant energy through advanced fuel cycles in the future (see Chapter 18), and therefore it can also be rightly considered as an asset worth a few trillion dollars. Therefore it is prudent to adopt a flexible approach. On the one hand, the disposal facility should safely isolate the irradiated fuel from the human population and the environment with minimal ongoing expense, and on the other hand, the disposal facility should permit its retrieval at a later date if appropriate and if so chosen.

IAEA [2012] provides guidance and recommendations for design, safe operation, and safety assessments of fuel storage facilities, both wet and dry.

4 Irradiated Fuel Storage and Disposal: Main Considerations

The main features that irradiated fuel storage and disposal facilities must incorporate are listed below. Other more detailed design and jurisdictional requirements exist, but are not covered in this book. The relative importance of a specific feature may differ from one facility to the next; therefore, an individual feature listed below may or may not be absolutely essential in a given facility.

(i) Avoid overheating of irradiated fuel

Even after discharge from the reactor, fuel continues to produce power, as explained in more detail in other chapters. Both the fuel and the materials used in storage and disposal structures and facilities have their respective temperature limits for safe operation. Therefore, the potential temperature increase caused by decay power must be controlled effectively to prevent overheating.

(ii) Limit chemical and metallurgical damage to irradiated fuel bundles

A variety of factors can damage stored and disposed fuel, e.g., chemical, metallurgical, and mechanical. Damaged fuel can release highly radioactive substances into the irradiated fuel bay and also potentially into the air above. Such radiological contamination must be controlled to acceptably low levels to protect the workers and the general public. Therefore, degradation and damage to bundles must also be limited to acceptably low levels.

(iii) Avoid mechanical damage to irradiated fuel bundles

Irradiated fuel bundles are handled remotely by fuelling machines (pre-programmed robots) during their removal from the reactor into the irradiated fuel pool, and by operators using remotely operated tools for subsequent transfers. These tools must be designed and operated so that the irradiated bundles are not damaged.

(iv) Avoid irradiated fuel configurations that could achieve criticality

CANDU fuel is made of natural uranium oxide and Zircaloy. Therefore, whether it is new or irradiated, it cannot be put into a configuration that will achieve criticality in ordinary water [Tsang, 1996]. This is not true, however, of LWR irradiated fuel, which contains higher levels of fissile isotopes than CANDU irradiated fuel.

(v) Protect workers and public from radiological exposure

Adequate shielding must be provided to protect the workers and public from the radiotoxicity of nuclear fuel. Whereas the cooling challenge reduces very quickly, shielding requirements last longer because some long-lived isotopes (particularly the transuranics) take thousands of years to decay. Nevertheless, the radioactivity of irradiated fuel does decrease with time, and therefore so does the radiological threat. Hence, the most demanding conditions exist when the fuel is handled during the initial wet and dry storage phases; special shielding must be designed to protect workers from radiation exposure.

(vi) Adequately safeguard irradiated fuel against proliferation

IAEA takes proactive measures to observe and record movements of nuclear fuel at reactor sites wherever practical. CANDU reactors are refuelled essentially continuously. This presents unique safeguards (tracking) challenges, particularly during discharge and initial storage of irradiated fuel in water-cooled bays. Continuous refuelling makes in-person monitoring impractical; hence, IAEA-approved monitoring by remote means is used. Several monitoring devices positioned at a number of locations in the reactor are used [Feiveson et al., 2011]. For example, neutron and gamma radiation detectors are used in the reactor vault and in the transfer port between the vault and the irradiated fuel bay (see Section 6.2); closed-circuit video cameras are used in the irradiated fuel bays; and tamper-indicating enclosures that have been inspected and sealed by IAEA are installed within the irradiated fuel bay.

(vii) Provide appropriate retrievability of irradiated fuel

Irradiated fuel may need to be retrieved from storage, especially after the fuel completes its designated storage periods in phases 1 and 2.

However, during phase 3—disposal/isolation—a nation may or may not choose to incorporate fuel retrievability into the design of its disposal facility, as explained in Section 3. Therefore, a reasonable balance must be struck regarding ease of access to irradiated fuel: access must be made quite difficult for a member of the general public, but not prohibitively so for an approved, legitimate, large, organized agency. The desired duration for such retrievability must also be chosen. These conscious choices influence the final configuration of the disposal facility. Canada's choice is described in a later section.

(viii) Avoid damage to facilities that interface with irradiated fuel

Similarly to the requirement for protection against damage to irradiated fuel, interfacing equipment must also be adequately protected, e.g., from corrosion.

5 Fuel Integrity

The above considerations promote safety during storage. Even though this section lists a number of items that focus on fuel integrity, several other aspects of storage also contribute significantly to overall safety; some of these are discussed in later sections.

As with safety of the reactor core, safety during storage and disposal must be confirmed during normal situations, during anticipated operational occurrences, and during postulated accidents. For brevity, this chapter focusses largely on normal situations. Broader discussions of safety during accidents are given elsewhere; see, for example, OPG [2009].

Recognizing that there will invariably be significant uncertainties in projections involving such long time periods, it is also common practice to vary a number of important parameters and assumptions and to perform bounding assessments. Furthermore, a number of hypothetical "what if" scenarios are usually also developed to explore the influence of parameters and scenario uncertainties in assessing long-term safety [NWMO, 2012a].

The remainder of this section provides some details on one aspect of safety during storage—fuel integrity.

5.1 Requirements

IAEA [1994] outlines, among other aspects, requirements for fuel integrity during storage. Lian [2010] cites the following two illustrative requirements from this IAEA document:

> "The spent fuel cladding shall be protected during storage against degradation that leads to gross ruptures, or the fuel shall otherwise be contained in such a manner that degradation of the fuel during storage will not pose operational problems" (Article 223),

and:

> "The heat removal capability shall be such that the temperature of all fuel (and fuel cladding) in a storage facility does not exceed the maximum temperature recommended or approved by the national nuclear regulatory body for the type and condition of fuel to be stored" (Article 225).

Furthermore, from IAEA [2002]: "The prediction of the integrity and retrievability of spent fuel constitute the main discussion topics for spent fuel behaviour regardless of the storage system and time period envisaged." Therefore, fuel integrity is an important consideration in the overall design of storage facilities.

As well, some fuel bundles do occasionally develop defects during operation in the reactor, i.e., before they are brought to storage facilities (see Chapters 17 and 18). Their pellets may well undergo faster degradation than those of intact fuel due to, for example, exposure of pellets to air, which would increase the rate of oxidation of UO_2. Therefore, the impact of fuel with defects on safety during storage must also be considered.

To quantify up-front the expected degradation of fuel in any given design or scenario, one must first identify credible damage mechanisms. Towards this end, the authors have culled from the literature some illustrative fuel damage mechanisms pertinent to storage and disposal and their key precursors and drivers. These are discussed in the next two sections.

5.2 Key Drivers and Precursors

<u>Decay Power</u>

As noted earlier, even after discharge from the reactor, nuclear fuel continues to produce power.

Standard methods are available to quantify decay power; for example, see Garland [1999], ANS [2014], and Glasstone [2014] for illustrative examples. Figure 1 shows how power decays with time in typical irradiated CANDU fuel. The figure shows power decay in a fuel bundle that produced 493 kW in the reactor just before its discharge. Immediately upon removal from the reactor core, an irradiated CANDU fuel bundle generates less than 10% of the power that it produced in the core. This figure drops to less than 1% in only a day after removal and to about 0.013% after a year has passed. The average power generated in a bundle at this point (one year) is less than 100 W, which is comparable to a household light bulb. This keeps dropping substantially with time, e.g., to about 5 W after 10 years and to about 1 W after 100 years, as shown in Figure 1. Hence, the decay power is small compared to the power produced by the fuel in the reactor; nevertheless, the fuel's temperature increase must be controlled effectively to avoid overheating the fuel and the storage structures.

<u>Fuel Temperature</u>

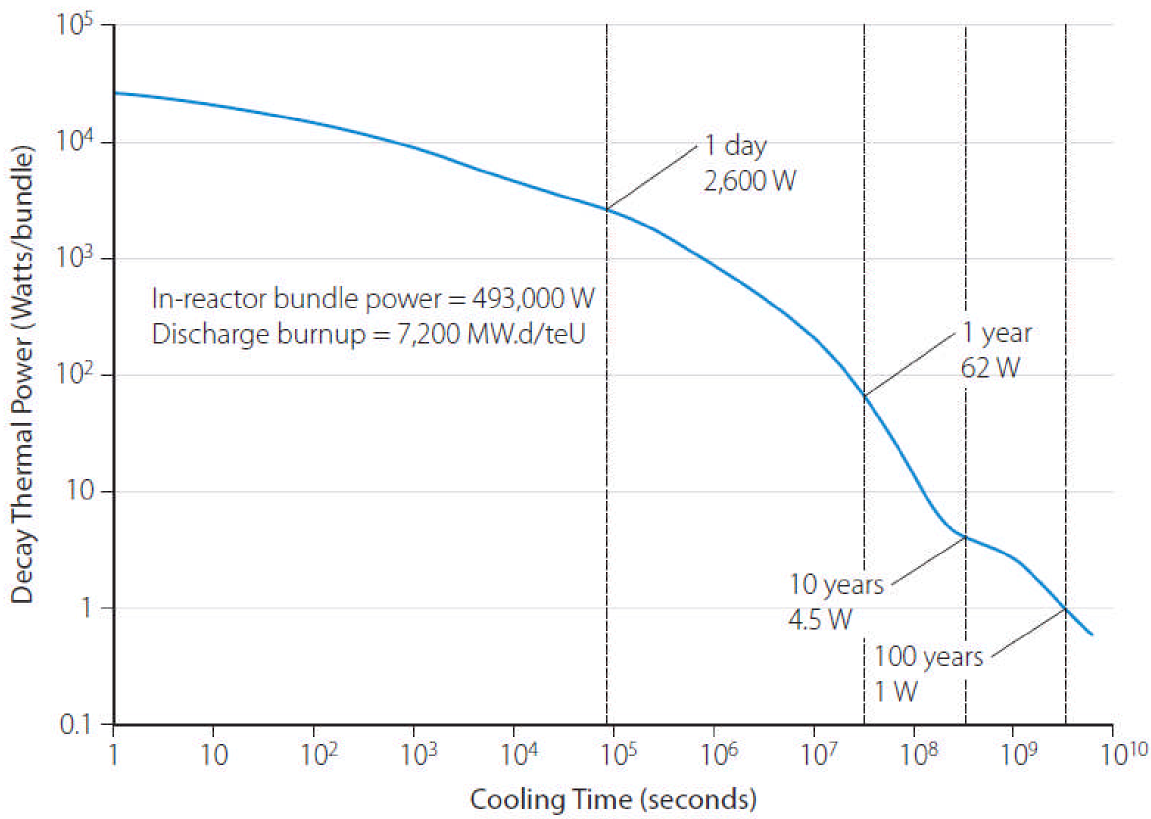
As an illustrative example of how fuel temperature depends on decay power, Figure 2 shows temperature decay in LWR[1] fuel and in its storage canister after emplacement in a long-term

Note 1: LWR means Light Water Reactor

repository for permanent disposal [Rothman, 1984]. In the repository, LWR fuel temperature peaks at about 330°C and then declines. For much of the latter half of the first millennium after emplacement, maximum LWR fuel temperature ranges from 90°C to 150°C. CANDU fuel temperature is expected to be comparatively lower due to its lower burn-up and therefore lower decay power.

Changes in UO_2 composition and microstructure

In the dry sealed environment inside containers, there are few processes that would significantly alter the local composition and microstructure of UO_2. Over long periods, however, some changes are likely to develop by processes such as ongoing decay, diffusion of radionuclides, and damage from alpha radiation.



[Illustrative Example; Courtesy Kwok Tsang]

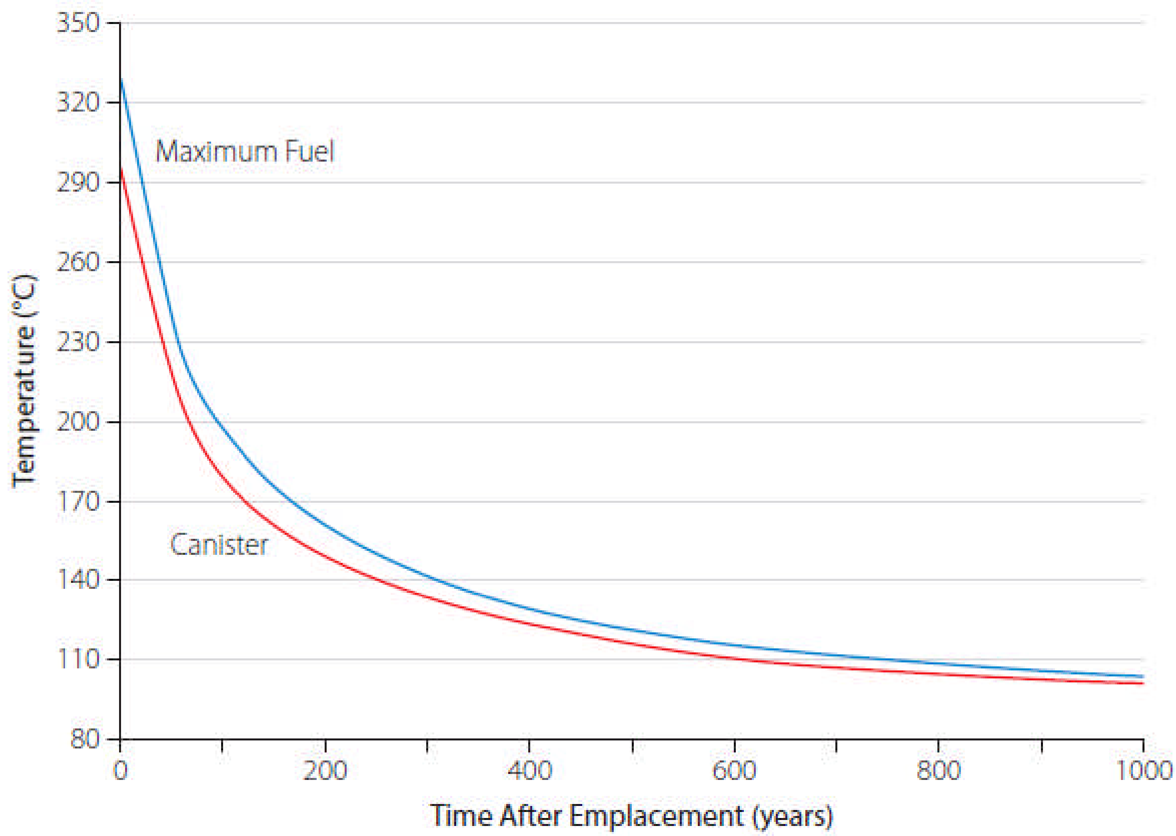
Figure 1 Decay power of an irradiated fuel bundle

Build-Up of Helium Gas and Internal Gas Pressure

As explained in Chapter 17, fuel elements contain: (a) initial filling gas that is added inside a fuel element during fabrication and (b) fission gas that is generated during irradiation. In addition, alpha decay during storage generates helium (He) gas. All three components together can create internal pressure that can frequently be higher than atmospheric pressure. The excess pressure (=internal minus external pressure) can create tensile stresses in the fuel element's

Zircaloy, which can persist over significant durations. If stress is excessive, even nominally slow mechanisms can potentially threaten fuel element integrity over the long periods pertinent to storage and disposal.

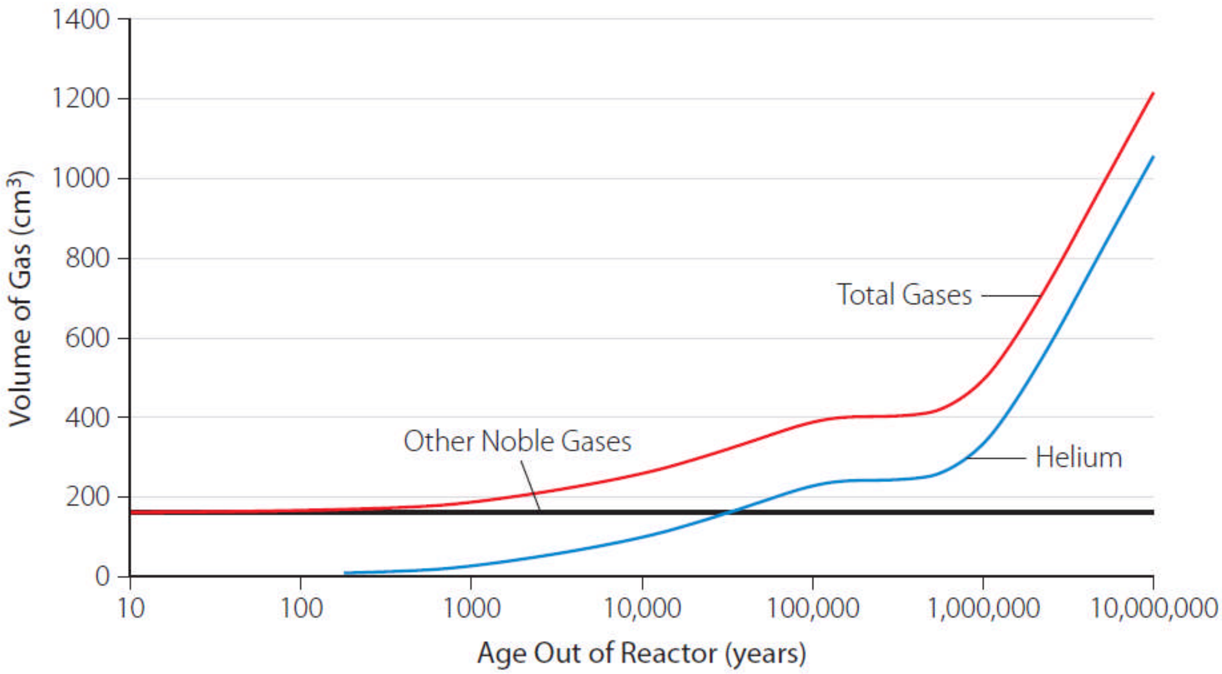
The following description has been reproduced largely from NWMO [2012a]. Helium is stable (i.e., not radioactive) and does not react chemically with other elements. Therefore, the total amount of helium gas in the fuel elements would be expected to increase with time during storage. Figure 3 gives an illustrative example, assuming that all fission-generated gases escape the UO_2 matrix into the open space inside the fuel element (see Chapter 17 for explanations of these terms). Fission gases are formed in the reactor, and therefore, except for radioactive decay, their amount would not change significantly during storage and disposal. In contrast, after about 30,000 years, the amount of helium would be equal to that of fission gases, so that the total amount of gas present would be double that under initial conditions. After about one million years, the rate at which helium is produced would slow down due to changes in the composition of the decay chain, but the total amount of helium within the fuel element would continue to increase.



[Source: Rothman, 1984]

Figure 2 Illustrative temperatures in an LWR fuel and canister

Figure 3 illustrates the amount of helium produced within grains of UO_2. Before it can contribute to internal gas pressure, this helium would need to reach the open void inside the fuel element. Pertinent processes that enable this are described in Chapter 17 and consist mainly of diffusion to grain boundaries, storage within grain boundary bubbles, inter-linkage of bubbles, and micro-cracking in grains. See Chapter 17 for more detailed descriptions of these phenomena.



[At normal temperature and pressure. Source: NWMO, 2012]

Figure 3 Gas volume inside a fuel element

5.3 Fuel Damage Mechanisms

Over extended periods of time, fuel temperatures generated as described above can lead to mechanisms for fuel damage during storage and disposal, as documented in a number of publications such as Rothman [1984], IAEA [2002], McMurry et al. [2003], Lian [2010], and NWMO [2012a]. Some illustrative damage mechanisms are summarized in the following paragraphs, mainly from NWMO [2012a].

Zircaloy Corrosion

Obviating excessive corrosion is a generic design requirement for fuel. This topic is addressed in Chapter 17 in the context of fuel residence in the reactor. In addition, during storage, sheath corrosion must be limited to acceptable levels.

Zircaloy is relatively resistant to pitting corrosion in the pure chloride-free water encountered in the fuel bay, and pitting does not occur in the atmosphere provided during dry storage.

Hydrogen Absorption and Zircaloy Embrittlement

As noted in Chapter 18, fuel sheaths can contain hydrogen and deuterium from several sources, mainly residual hydrogen produced during fabrication, hydrogen or deuterium picked up from the coolant during irradiation, and hydrogen or deuterium generated by in-reactor corrosion of the sheath. Still more hydrogen can enter the Zircaloy as a result of radiolysis in the storage pool; this can be limited by chemistry control and by purification of pool water. If quantities are excessive, some hydrogen and deuterium may precipitate in the Zircaloy as zirconium hydride or deuteride, especially at the relatively cooler temperatures of storage after removal from the reactor. These hydrides and deuterides result in more brittle (less ductile) Zircaloy that is more susceptible to fracturing from the mechanical loads imposed on it, for example during fuel transfers and/or during fuel extraction from or loading into a disposal container.

Delayed Hydride Cracking (DHC)

DHC is similar to the process mentioned above (embrittlement of Zircaloy caused by hydrogen), but occurs after a time delay, as explained below.

In some situations, this hydrogen or deuterium may be distributed fairly uniformly. In such situations, their local concentrations could be acceptably low. However, if the material is under significant temperature and/or stress gradients for a significant duration, hydrogen and deuterium can migrate and concentrate preferentially at locations of relatively lower temperature and higher stress. Over time, this can potentially lead to excessive local hydride and deuteride concentrations and therefore to higher susceptibility to brittle fracture.

Sheath Creep and Rupture

Internal gas pressure can lead to long-term creep of Zircaloy. If excessive, this can potentially rupture the Zircaloy.

Stress Corrosion Cracking (SCC)

As noted in Chapter 17, SCC (also called environmentally-assisted cracking, or EAC, in Chapter 17) occurs when irradiation-embrittled Zircaloy experiences sufficiently high tensile stresses in the presence of a corrosive environment for a sufficiently long duration. Depending on the magnitude of stress and of intensity of the corrosive environment, through-wall cracks due to SCC can occur quickly, or slowly, or not at all. During storage, long-term stresses are provided by internal gas pressure. A corrosive internal environment is provided by fission gases that were released during irradiation in the reactor.

Mechanical Overstress

From NWMO [2012a]: "As long as the fuel bundles are supported by baskets in intact containers, they are not subjected to significant load-bearing stresses. If tremors associated with earthquakes caused the fuel bundles to vibrate sufficiently, presumably some of the fuel pellets or the cladding could be damaged. The damaged material would remain in an intact container, and the overall evolution of the used fuel bundles would not be significantly changed."

Fatigue

This would be a consideration mainly during transportation of irradiated fuel bundles, e.g., to sites for dry storage and/or long-term disposal or isolation.

Defective Fuel

Sometimes fuel can develop defects in the reactor, that is, the Zircaloy develops a hole or a crack. Detailed discussions of this phenomenon are provided in Chapters 17 and 18.

A hole or crack in Zircaloy will allow oxygen-containing air to come in contact with UO_2. This, over time, can potentially oxidize the pellet to higher states of oxygen, such as U_3O_8. The latter is less dense than the former, and therefore the pellet can swell. Excessive swelling of the pellet can potentially split the sheath. Under these conditions, in addition to the original hole or crack, the Zircaloy can develop further splits during storage.

Other Effects and Remarks

The literature also mentions other mechanisms pertinent to potential fuel damage, for example, hydraulic processes and biological processes. They are not covered in this chapter for the sake of brevity. Combinations of these various mechanisms are also possible.

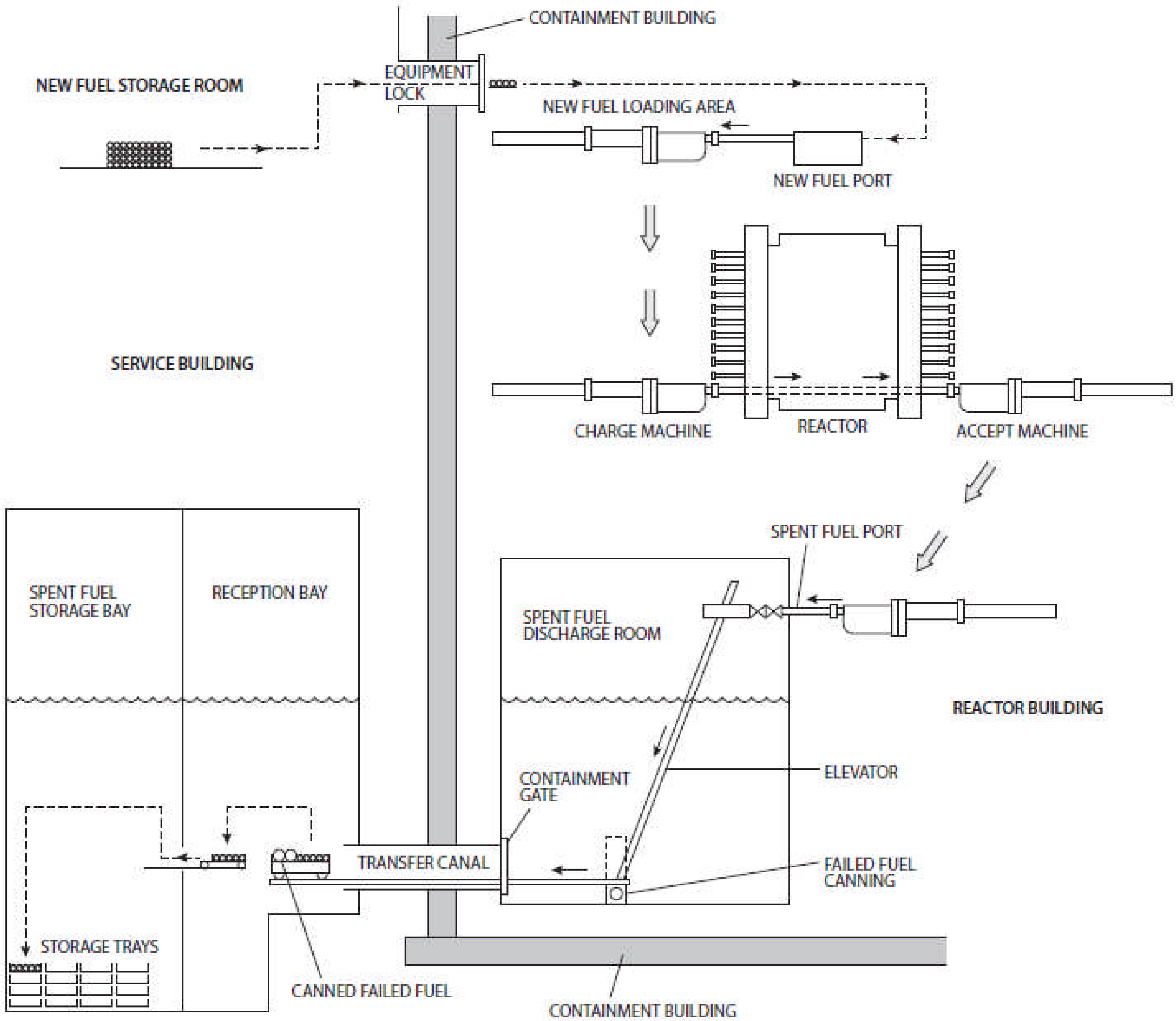
These fuel failure mechanisms during storage have been postulated partly from first principles and partly from exploratory experiments. Because several of these mechanisms are thermally activated and increase non-linearly with temperature, it would be good design practice to keep storage temperatures reasonably low. Design studies as well as experience to date indicate that under realistic as-designed operating conditions, the actual risk of fuel failure from any of these postulated mechanisms is negligibly small. Some of these aspects are illustrated in the following sections.

6 Wet Storage at Reactor Sites

Storage of irradiated fuel entails engineered facilities that can and must be monitored for the entire time that the fuel is stored and that make retrieval and movement of irradiated fuel to other facilities readily feasible.

6.1 Flow of Fuel Bundles Through a CANDU

Because CANDUs use natural uranium fuel, they need to be refuelled essentially continuously to maintain the required level of reactivity within the core. From practical considerations of work scheduling and equipment maintenance, refuelling is not done continuously, but rather during parts of each day. The flow of fuel through a CANDU is shown in Figure 4.



[Courtesy Ralph Granz]

Figure 4 Flow of fuel through a CANDU reactor

Inside the core, fuel bundles reside in horizontal fuel channels and are cooled by high-pressure heavy water. During refuelling, new fuel bundles are inserted into a fuel channel (usually in the same direction as the coolant flow), and irradiated bundles are removed from the same fuel channel by two fuelling machines operating in tandem. One machine, labelled as the “charge” machine in the figure, inserts new fuel bundles at one end of the channel, and the other, labelled the “accept” machine, removes irradiated bundles from the other end. The two machines reverse charge/accept roles depending on the channel being fuelled (i.e., the flow direction in the channel being fuelled). Fuelling machines, when connected to the fuel channel, act as extensions of the fuel channel pressure boundary and provide the required cooling for irradiated fuel bundles while they are inside the fuelling machine. The “accept” machine delivers the irradiated bundles to the irradiated fuel (spent fuel) port. From there, the bundles are moved by other remotely operated means to the irradiated fuel discharge room, which contains ordinary water at atmospheric pressure.

During their residence in the reactor, and during all movements within the reactor and the irradiated fuel storage bay, the bundles are oriented in the horizontal position. This is the orientation for which the bundles are designed when they are subjected to the flow and power conditions inside the reactor fuel channel (Chapter 17), and therefore this would be the preferred orientation for the bundles throughout their storage and disposal life. However, a single tier of vertical bundles supported on their end plates is also an acceptable configuration, provided that the bundles are not required to support more than their own weight.

6.2 Irradiated Fuel Bundles Temporarily in Air

At the spent fuel port (Figure 4), the irradiated fuel bundles transit from the fuelling machine, which contains heavy water, through an air-cooled interlock and elevator at atmospheric pressure, and are then deposited in the spent fuel discharge room, which contains ordinary (light) water. CANDU designs that use enriched fuel and ordinary water for coolant, such as the ACR [AECL, 2007], obviate the passage of irradiated fuel through air, and the transfer of the bundles can be performed underwater from the fuelling machine to the spent fuel discharge room (or possibly directly to the reception bay). In existing CANDUs, movement of bundles through the air interlock and elevator is timed so that irradiated bundles do not overheat while in air.

As an illustrative exercise, you are encouraged to determine the maximum length of time that the bundles can safely stay in air. Start with the information on decay heat given in Section 5. Assume power equivalent to zero time after discharge and no heat removal by air. Use the standard equation for adiabatic heat-up and apply it to a fuel element. How long does it take for the sheath temperature to reach the melting point of Zircaloy?

CANDU plants are equipped with water spray nozzles to cool the bundles if their residence in air exceeds the maximum permissible time.

6.3 Throughput of Irradiated Fuel Bundles

From the spent fuel discharge room (Figure 4), irradiated bundles are moved to the spent fuel reception bay through the water-filled transfer canal. From the reception bay, the bundles are moved to the spent fuel storage bay, where they are loaded underwater by a remotely controlled bundle loading apparatus into storage trays that rest on storage racks. More recently, consideration has been given to replacing the storage trays or racks with storage baskets which are directly transferrable to dry storage containers. Both the trays or racks and the baskets are designed to ensure:

- adequate cooling of the bundles in the storage bay,
- avoidance of damage to bundles during their residence in the pool,
- avoidance of damage to pool components,
- compliance with safeguards monitoring requirements (e.g., bundle serial numbers are readily visible),
- direct transfer of bundles (baskets only) into dry storage containers, and
- use of the most compact arrangements of bundles in the pool.

As an illustrative exercise, one can determine the number of bundles discharged from a CANDU 6 reactor in one year, assuming the following:

- The plant's electrical output is 600 MW(e).
- The plant overall efficiency of conversion of thermal power to electricity is ~30%.
- Capacity factor (fraction of time the plant is running at full power) is 90%.
- Each bundle contains approximately 19 kg of U and produces thermal power (average discharge burn-up) of 6,500 MWd(th)/teU.

Using the above parameters, one can determine:

- Total thermal energy produced in one year: 657,000 MWd(th) (=600x0.9x365/0.3).
- Average thermal energy produced by an individual fuel bundle: 123.5 MWd (th) (=6500x19/1,000).
- Number of bundles required to produce the reactor's annual power: 5,320 bundles (=657,000/123.5).

The storage pool is designed to accommodate the spent fuel generated from 10 years of operation (~53,200 bundles in this example) plus one full core-load of bundles (4560 bundles in this example; 380 channels containing 12 bundles each). The additional capacity of one core-load provides for the possibility of having to unload the entire core to accommodate refurbishment. In practice, bundles could begin to be removed from the pool as early as seven years after their placement into the pool, reducing the risk that pool capacity would be reached anytime during reactor life.

6.4 Irradiated Fuel Bay: Typical Dimensions

In Canada, there are several locations where irradiated fuel is either currently stored in fuel bays or has been stored in the past. Some of these locations are: Chalk River Laboratories (CRL) at Chalk River, ON; Whiteshell Laboratories (WL) at Pinawa MN; McMaster Nuclear Research Reactor, Hamilton, ON; Point Lepreau Nuclear Power Plant at Point Lepreau, NB; Gentilly

Nuclear Power Plant at Trois-Rivières, QC; Darlington Nuclear Power Plant in Darlington ON; Pickering Power Plant at Pickering ON; and Bruce Power Plant at Tiverton ON.

Designs, layouts, and dimensions of irradiated fuel bays (IFBs) differ from one station to another. As an illustrative example, the following discussion concerns an IFB with three sections: a reception bay; a long-term (or main) storage bay; and a transfer bay.

Irradiated fuel is received in the reception bay and stored there for one to two weeks (Figure 4). Then it is transferred to the long-term storage bay and stored there for a few years. The long-term storage bay usually also has a fuel inspection station where fuel can be inspected underwater, for example to monitor and assess fuel performance. Finally, in the transfer bay (not shown in Figure 4), fuel is loaded (underwater) into storage baskets en route to placement in dry storage

Figure 5 illustrates a typical irradiated fuel storage bay. The sizes of the bays differ in different stations, e.g., as small as 20 m x 12 m or as large as 34 m x 17 m [OECD/NEA, 2015]. These have a surface area similar in size to an Olympic-size swimming pool, but the IFBs are constructed of double-walled reinforced concrete and are much deeper. To provide sufficient radiological shielding to workers above, water in IFBs is usually 6–8 m deep. Typically, the volumes of water in C6 IFBs are: reception bay, 700 m^3, and main storage bay, 2,000 m^3.

6.5 Irradiated Fuel Bay: Heat Removal

Heat must be removed from the fuel bay to maintain water at a predetermined temperature and to continue to provide cooling to the fuel. Generally, an irradiated fuel bay with 10-year storage is maintained at <38°C; with 10-year storage plus one full core load, the pool is maintained at <49°C. Major elements of the cooling system are a pump, a heat exchanger, and a resin bed.

As industrial installations go, incorporating the required heat removal capability into the irradiated fuel pool cooling system is not a significant challenge for the designer. However, in case of impaired heat removal capability, it is important for safety reasons to determine how much time may be allowed to elapse before auxiliary cooling must be provided. In the example above, the reader is encouraged to determine how much time is required for the cooling water to evaporate and the fuel bundles to be exposed to air, and how that time changes with the number of bundles in the bay.

In the post-Fukushima era, factors such as available response time based on pool size and the availability of off-site emergency cooling have achieved heightened prominence.

[Source: Villagran, 2014]

Figure 5 A typical irradiated fuel bay

6.6 Corrosion of Irradiated Fuel in the Pool

Corrosion of CANDU fuel during its residence in the reactor is covered in Chapter 18. To in-reactor corrosion, any additional corrosion occurring during the residence of the fuel in storage pools must be added. To help reduce the latter, demineralized water is used in the pool.

Because CANDU fuel's residence time in the reactor is relatively short (especially compared to LWR fuel), the resulting in-reactor corrosion is very low (see Chapter 18). Experience to date has been that additional corrosion during storage is trivial compared to the corrosion picked up during residence in the reactor. Hence, the total corrosion in-reactor and during pool storage is very small and does not significantly reduce the fuel bundle sheath thickness.

6.7 Shielding of Irradiated Fuel in the Pool

For shielding to protect workers and the public, irradiated fuel storage pools typically contain about 7 m of water above the last rack of fuel.

6.8 Criticality Not an Issue for CANDU Irradiated Fuel

The IAEA (and CNSC) design guidelines for irradiated fuel storage and disposal include a need to assess conditions that could lead to criticality. There are no criticality-based restrictions on proximity of irradiated CANDU bundles in the pool because the remaining fissile content in irradiated bundles is low. Bundles can be placed into the most compact configuration (based on heat transfer considerations only) and do not need to be "re-racked" during their residence in the pool.

7 Dry Storage at Reactor Sites

As noted earlier, after a few years in water pools, fuel bundles are transferred to dry storage because it is a comparatively less expensive method of storing large quantities of fuel and because after a few years out of the reactor, fuel requires much less cooling.

A variety of options are available for dry storage. As one illustrative example, *dry storage containers* are used at the Pickering and Bruce plants [OPG, 2015]. As another illustrative example, Atomic Energy of Canada Limited (AECL) has developed *concrete canisters* that are used at some C6 plants and decommissioned prototype plants. These are described in the following sections.

7.1 Dry Storage Containers

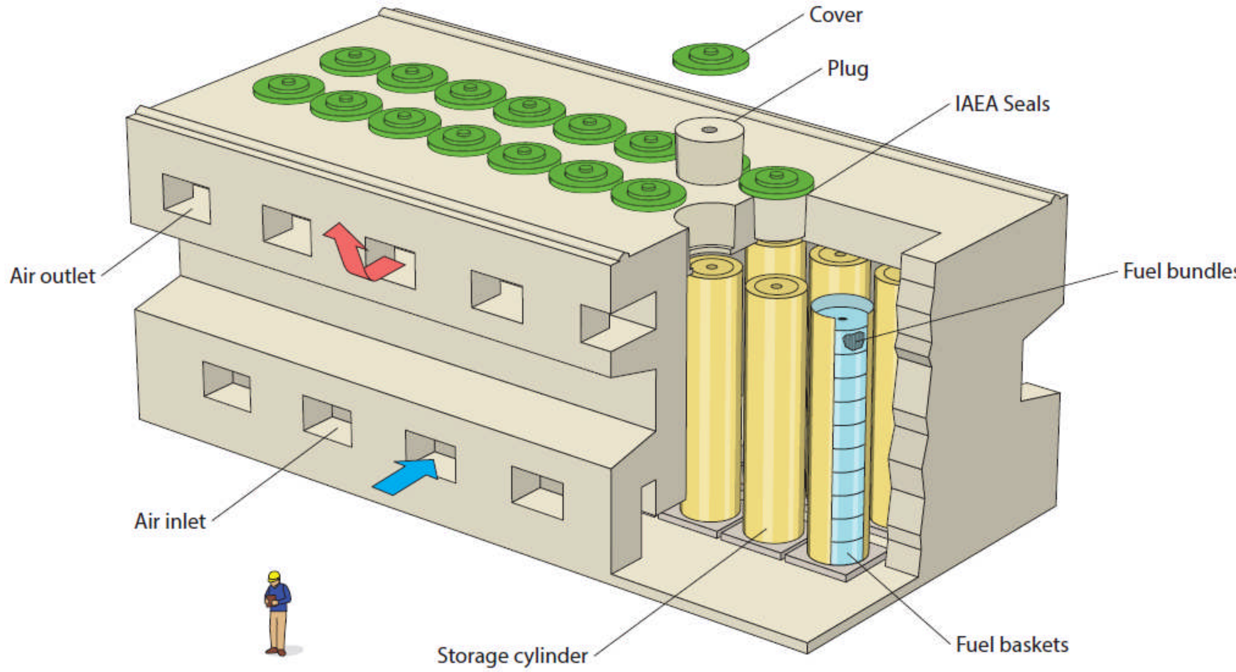
Each dry storage container (DSC) can store a few hundred fuel bundles [OPG, 2015]. It is made of reinforced high-density concrete approximately 51 cm thick and is lined inside and outside with 12.7 mm-thick steel plate. The thick concrete provides an effective barrier against radiation. Irradiated fuel bundles are loaded into a DSC within the IFB. The DSC is then sealed, dried, and moved indoor into a secure storage building.

7.2 Concrete Canisters

AECL has developed modular "concrete canisters" such as CANSTOR and MACSTOR [CANDU Energy, 2014]. The two are similar in concept, but MACSTOR is significantly bigger. A brief overview of the latter is provided here; Figure 6 illustrates its principal features.

In the MACSTOR design, many fuel bundles are stored in a "storage basket"; many baskets are in turn placed in a steel "storage cylinder". A large number of storage cylinders are inserted into a concrete labyrinth called MACSTOR. Shielding is provided by the concrete of the module, and heat removal is achieved by air circulation due to natural convection. In Figure 6, note the staggered openings at the bottom and top of the module; they allow air circulation, but do not provide a direct line of sight through the concrete to the storage cylinder. Three barriers prevent escape of radionuclides from the fuel: the fuel sheath, the sealed basket, and the storage cylinders. Irradiated bundles can be retrieved using appropriate shielded equipment.

The cross section of each module is about 8 m wide and about 7 m high. Each MACSTOR unit can store about 12,000 CANDU fuel bundles.

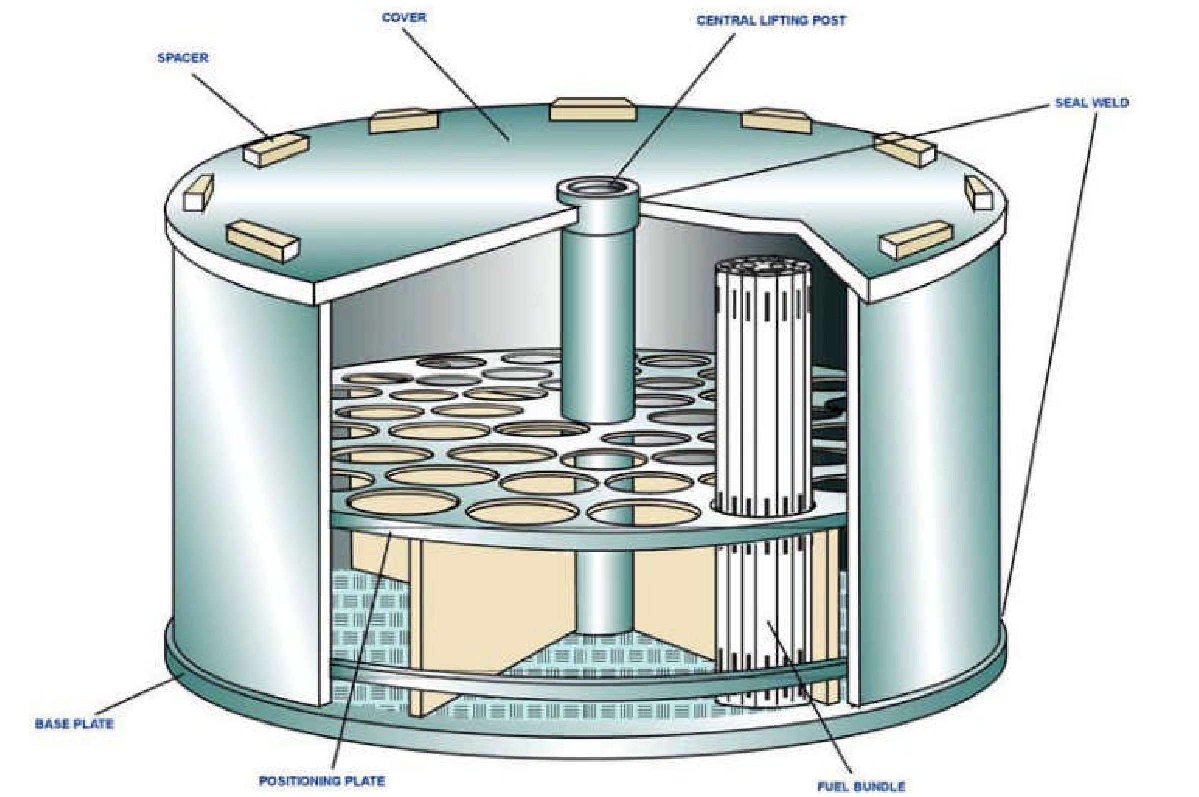


[Courtesy Jim Lian]

Figure 6 Overview of MACSTOR

7.3 Storage Baskets

Figure 7 [OPG, 2009] illustrates the main features of a typical storage basket used in MACSTOR. Fuel bundles are loaded into baskets underwater in the fuel transfer bay, dried and weld sealed inside the baskets, then moved in shielded flasks and loaded into storage cylinders. Bundle orientation is changed from the normal horizontal orientation in the reactor and the irradiated fuel storage bay to the vertical orientation during the basket loading process. In this design, the basket is typically designed to hold 60 CANDU fuel bundles oriented vertically. Ten such baskets are loaded into a metal storage cylinder.



[Source: OPG, 2009]

Figure 7 Basket for dry storage of fuel

7.4 Transfer of Bundles from the Pool into Baskets

Shielding requirements dictate that bundles must be loaded into dry storage baskets underwater using remotely operated equipment. Design and process details differ from one station to another.

7.5 Heat Transfer

MACSTOR facilities enable dry storage of fuel bundles with up to about 6 W of power per bundle. This is typical of C6 fuel that has been cooled for six years.

Considering the size and good heat transfer configuration of the fuel elements, it can be easily recognized that effective dissipation of this amount of heat from CANDU fuel can be achieved by heat transfer through convection, conduction, and/or radiation without any parts of the element or the bundle becoming overheated. Ultimately, the heat is passively transferred from the storage cylinder to the atmosphere through natural convection.

7.6 Fuel Integrity

For dry storage, the target criterion is to limit sheath failures to within 1% of fuel rods during 100 years of storage [Lian, 2010].

Lian [2010] provides an overview of the integrity of irradiated CANDU fuel during interim dry storage in MACSTOR and the associated research and development. Lian concludes the following:

- Highest fuel temperature is less than 150°C; and
- Maximum sheath hoop stress is less than 4 MPa.

The above conditions are less demanding than those experienced by LWR fuel during dry storage. Therefore, based on detailed assessments of LWR fuels done by the Electric Power Research Institute in the United States, Lian [2010] concluded the following for CANDU fuel stored in MACSTOR:

- Creep-rupture and external oxidation should not cause failure at storage below 300°C;
- For storage temperatures below 300°C, stress-corrosion cracking (SCC) is the most likely mode of fuel failure, but even it is unlikely to occur to any significant extent during dry storage;
- Fatigue is not a limiting failure mechanism for stored fuel sheaths; and
- Sheath splitting by UO_2 oxidation is a limiting mode only for fuel rods that have already developed defects.

Based on the above discussion, Lian [2010] projects that the failure rate in CANDU fuel during dry storage in MACSTOR would likely be in the neighbourhood of 0.001% of fuel rods in 100 years. This is 1,000 times lower than the criterion noted earlier.

7.7 Shielding

As also noted in Section 5, radiation (i.e., the shielding requirement) does not diminish proportionately as much as heat generation and requires continuing attention in handling of irradiated fuel. As an illustrative example, nominal wall thickness in MACSTOR is approximately one metre of ordinary concrete to control direct radiation exposure at the outside wall. This is an indication that significant radiation can be expected from stacks of irradiated fuel bundles, even six years after discharge.

8 Final Disposal or Isolation

As of 2014, Canada had about 2.5 million irradiated fuel bundles in storage [Garamszeghy, 2014]. Following the two interim phases of irradiated fuel storage described above, the irradiated fuel may be either reprocessed or placed into permanent disposal or isolation facilities. This section touches upon the following aspects of Canada's current concept for permanent disposal or isolation of irradiated fuel: approach and strategy; repository design; container design; and health protection.

The remainder of this section has borrowed heavily from lectures delivered by J.E. Villagran at a course sponsored by the Canadian Nuclear Society [Villagran, 2013, 2014].

8.1 Strategy

Several options are available for permanent disposal or isolation of irradiated nuclear fuel, including isolation in stable geological formations or emplacement in abandoned salt mines. After many decades of research [Boulton, 1978], detailed evaluations, and public consultations [Seaborn, 1998], Canada has opted for centralized containment and isolation of irradiated nuclear fuel in a deep geological repository (DGR) [NWMO, 2015a]. Several other nuclear countries are also inclined towards geological repositories, e.g., Finland, Sweden, France, Germany, Japan, Switzerland, and the United Kingdom.

> "The goal of (Canada's) plan is to place the (irradiated) fuel deep underground in rock where it will be constantly watched to make sure it is secure and not affecting anything around it. Then at some point in the future, people can decide if they want to close the facility and return the ground to its natural state" [NWMO, 2015b]. This approach includes an optional step of interim shallow underground storage.

A key aspect of the current concept is to surround the irradiated fuel by multiple barriers, as explained later in this section. This minimizes the potential for release of radionuclides should any individual barrier fail.

The Canadian strategy for long-term waste fuel management is called "adaptive phased management" (APM) [NWMO, 2015a] and is being administered by the Nuclear Waste Management Organization (NWMO). APM covers all phases of high-level waste management, including dry storage of fuel at the reactor site; the end result is final disposal (e.g., isolation in a DGR).

8.2 Phases of Disposal

One can define two distinct phases in the life of a disposal or isolation facility:

- The initial, operational period during which the facility is constructed, filled with fuel, and closed; and
- The long period of isolation of irradiated fuel after closure of the facility.

Initial operational period: Canada's strategy and design preserves the retrievability option and provides for continuous monitoring. Thus, APM engenders flexibility in design and promotes ongoing technical and sociological research, enabling continuous learning and adaptation. The process is collaborative, open, inclusive, and transparent, and decision-making is phased. The current expectation is that this phase will last a few decades.

Isolation period after closing the facility: The facility's closure is a few decades away, and society's needs and preferences that far away cannot be reasonably anticipated at this time with sufficient certainty. Therefore, it is best to be flexible for now about the facility's configuration after closure. NWMO's intention is to "remove underground equipment and backfill and seal the access tunnels and shafts. Surface facilities will also be dismantled at a pace and in a manner determined collaboratively with the community, regulators, and other interested individuals" [NWMO, 2012b]. In this sense, there is an option, even an intention, eventually to "entomb" the fuel.

From "first principles" many factors can be postulated as important for detailed quantitative assessment of repository and container design. The following sub-sections highlight a few such factors. Experience and actual analyzes have shown that some are relatively more important than others.

8.3 Repository

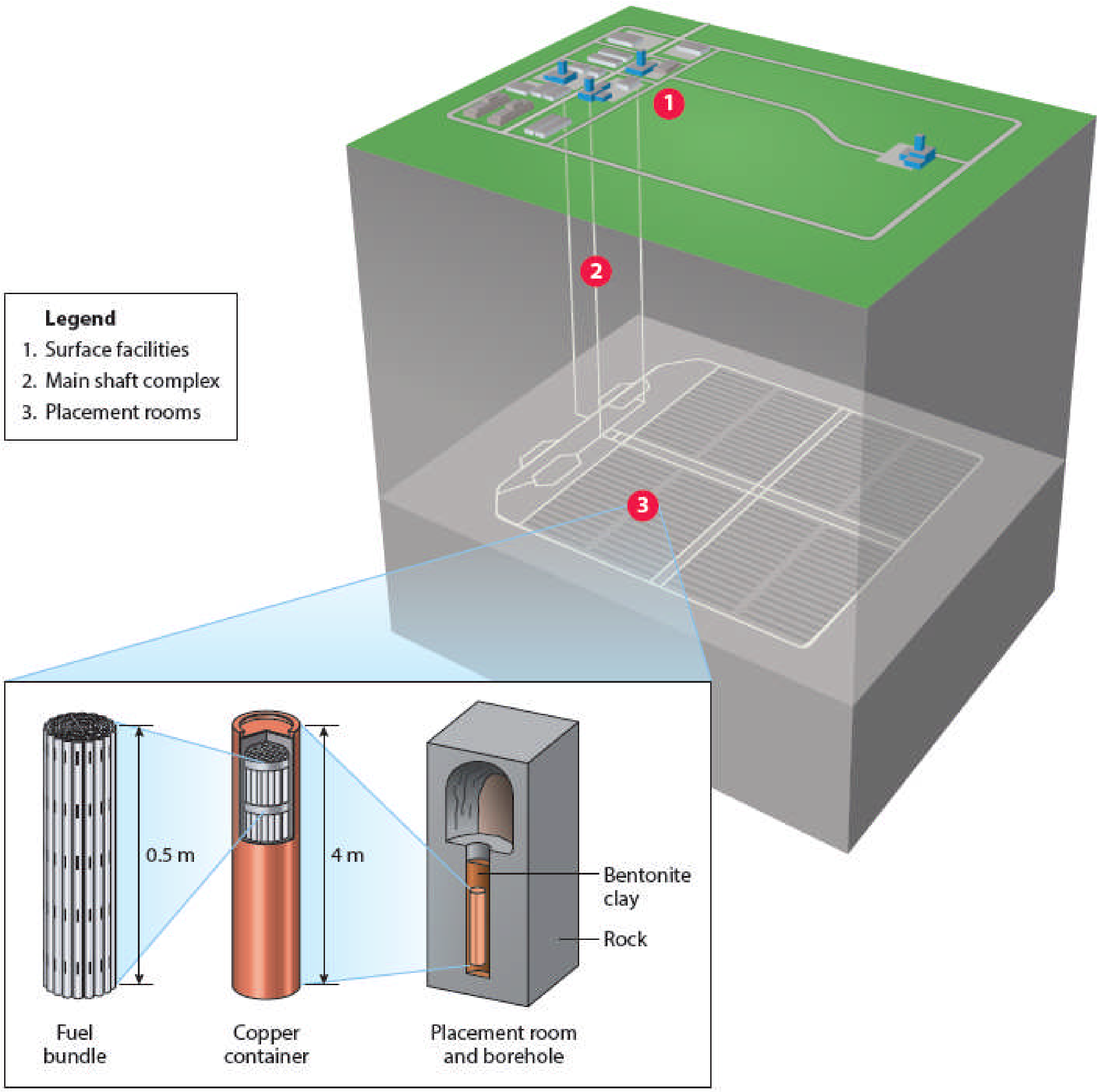
Figure 8 shows a conceptual design of an illustrative repository. The overall concept starts with placing many fuel bundles inside a sealed, corrosion-resistant container called a "used fuel container" (UFC). This step isolates the fuel bundles from groundwater and delays release of radionuclides into the environment.

The containers in turn are surrounded by a buffer material, bentonite clay, which possesses two physical characteristics that enhance its function of isolating the fuel. First, this natural clay can absorb up to 10 times its weight in water. If moisture were to approach the containers filled with used fuel, the bentonite clay would absorb the water and swell up, forming a seal around the container. This would delay the groundwater from reaching the container. Second, this clay has low permeability. Therefore, should the container fail, the clay would slow the diffusion of radionuclides out of the repository.

Many such containers would be placed in a horizontal position inside placement rooms in the repository, at a depth of some 500 m. Other concepts are also being considered, such as vertical placement of containers in boreholes.

Thus, multiple barriers isolate the fission products from the biosphere: the UO2 matrix; the Zircaloy sheath; the sealed copper containers; bentonite clay; and rock.

Current projections are that by their end of life, CANDU power reactors will irradiate some 4.6 million fuel bundles. To dispose of these, a typical repository would require an area of about 2 km x 3 km, or about 600 Ha or 1,480 acres.



[Illustrative concept; Source: Villagran, 2014]

Figure 8 Concept of a deep geological repository

8.4 Used Fuel Containers

This section illustrates the main features of two designs of used fuel containers (UFCs) that use a copper shell.

8.4.1 Conceptual designs

Figure 9 shows two slightly different container concepts; each container can hold many CANDU fuel bundles. In the first concept, an important barrier to corrosion is provided by a shell made from oxygen-free phosphorus-doped (OFP) copper 25 mm thick. Structural support is provided by an inner liner made of carbon steel 96 mm thick. In the second concept, the copper shell is replaced by a copper coating about 3 mm thick [Hatton, 2015].

The number of UFCs in a typical repository depends on the detailed design of the UFC. For example, about 100,000 UFCs of Concept # 2 (as shown in Figure 9) would be needed to accommodate some 4.6 million fuel bundles.

8.4.2 Container lifetime

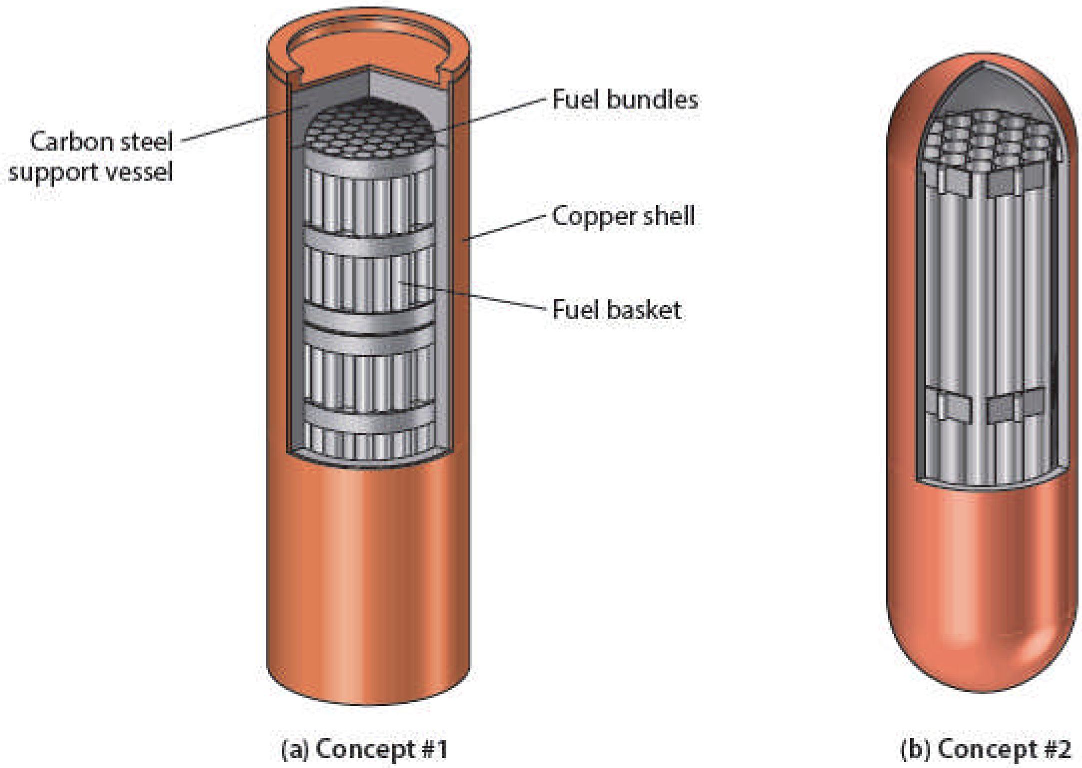
The major damage mechanisms are expected to be uniform corrosion and pitting of the copper shell or coating. Corrosion in turn can be driven by two major environmental factors [Villagran, 2014]:

- By initially trapped oxygen: the resulting corrosion is equivalent to about 0.080 mm of wall penetration;
- By chloride and/or sulphide from groundwater, from pyrite impurities in the clay, or by microbial activities. These cumulative corrosive interactions can cause additional wall thickness loss of about 0.27 mm over a period of one million years.

After increasing nominal estimates by applying conservative factors, the total potential loss of container wall thickness is estimated to be about 1.7 mm over one million years. Therefore, depending on the thickness of copper, a typical UFC should stay intact for over one million years.

8.5 Buffer Material

Main design requirements for the buffer material (bentonite clay in the above example) include preventing groundwater migration to the disposal container, slowing migration of radionuclides away from the container, and maintaining the long-term integrity of the chemical and mechanical properties of clay.



[Source: Villagran, 2013, 2014]

Figure 9 Containers for Permanent Disposal of Irradiated Fuel

8.6 Rock Temperature

The power generated by the fuel mass will increase its temperature, which will cause heat to be transferred from the fuel to the surrounding rock mass. Therefore, the temperature of the disposal vault is a design criterion for a repository. For hard-rock (granite, sedimentary rock, clay) geological disposal facilities, the temperature must remain below 90°C–100°C, but for salt repositories (such as the Asse facility in Germany or the WIPP facility in New Mexico), a higher temperature of up to 250°C may be acceptable. The higher-temperature facilities are more appropriate for disposal of reprocessed waste, which has a higher short-term heat load.

Consequently, the thermal calculations revolve around geometry: how much fuel can be placed into each container and how close to each other the containers can be placed to maintain a temperature below the design requirement. The limiting parameters for heat transfer to the rock vault are usually the surface temperatures of the container and the vault, rather than the maximum fuel temperature in the container.

Initial calculations indicate that it takes between 25 and 40 years for the temperature of a fully filled vault to go from ambient to 90°C, which would be a rather high temperature for workers. Hence, one might need to consider potential arrangements for cooling the vault during the waste emplacement period.

The rock temperature will be higher near the fuel and lower further away. This will lead to non-uniform local expansion of the rock, which in turn will lead to thermal stresses. From "first principles", one can postulate that if the temperature gradients and the resulting thermal stresses are large, the rock can potentially crack locally. However, are the thermal gradients expected to be large enough to crack the rock?

As an illustrative example, let us assume that the rock temperature inside the repository is 100°C. The minimum temperature in the Earth's crust at the depth of the repository must be greater than the freezing temperature of water (because otherwise leaching and migration would be impossible). Therefore, the maximum temperature gradient in the rock is ~100°C. Considering that it takes at least 25 years for the vault temperature to increase by 70°C (see the discussion earlier) provides a strong indication that the heat flux into the rock is extremely small and therefore that the local stress anywhere in the rock due to the temperature gradient is also very small.

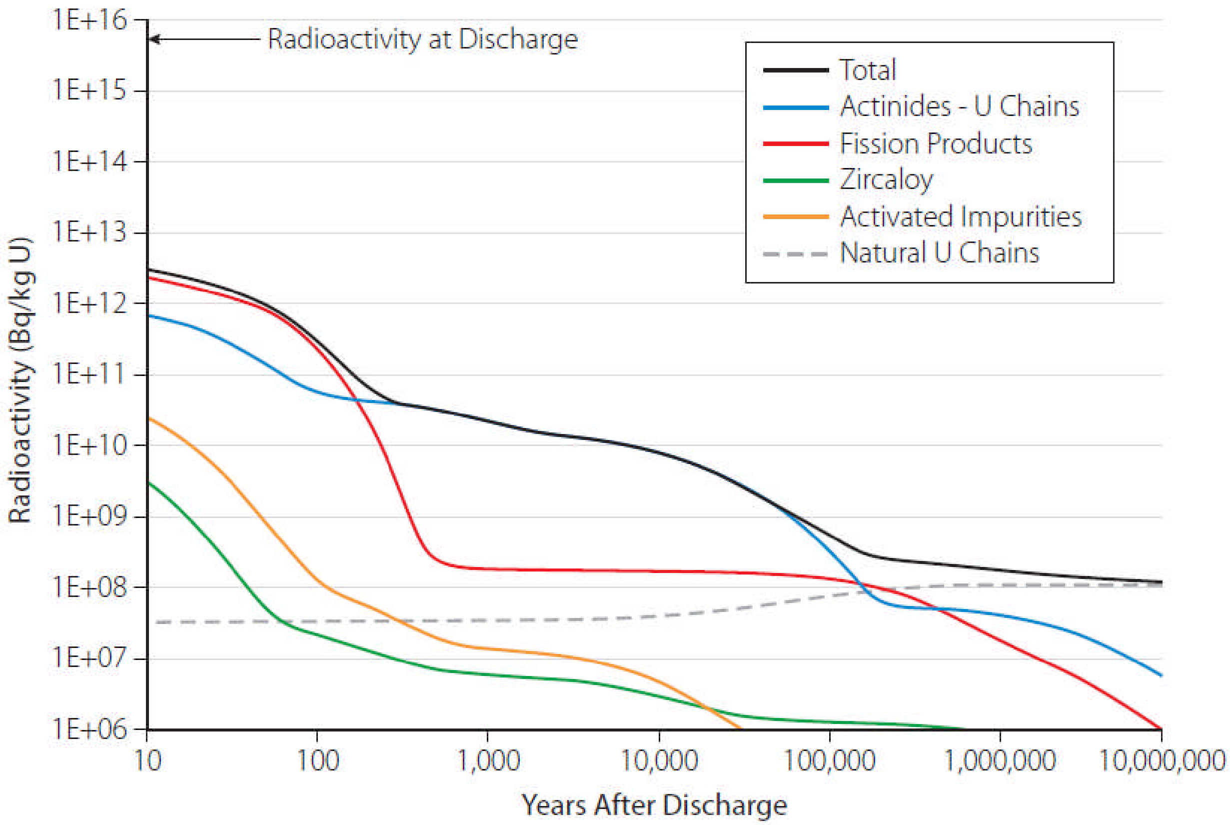
8.7 Health Protection

8.7.1 Radioactivity and relative toxicity

Figure 10 [Boulton, 1978] illustrates the typical decay of radioactivity of irradiated nuclear fuel for up to ten million years after discharge. Shortly after 100,000 years, the radioactivity of irradiated fuel is very similar to that of natural uranium plus its decay chain.

The only credible escape route of radioactivity from disposed fuel to the biosphere is by transport through groundwater. Therefore, an illustrative perspective on the potential hazard of nuclear waste can be obtained by comparing its toxicity in water with that of naturally occurring materials.

The International Commission on Radiological Protection (ICRP) has defined a "relative toxicity index" as the ratio of the volume of water needed to dilute the radionuclides in waste material to drinking water standards to the volume of water needed to dilute an equal weight of 0.2 percent uranium ore to the same standards [Boulton, 1978]. Figure 11 shows some illustrative results of these relative toxicity calculations [Boulton, 1978]. The calculations necessarily contain some assumptions, but nevertheless, for purposes of an initial illustration, the figure suggests that after about a century following removal from the reactor, the relative toxicity of used nuclear fuel (in groundwater) is similar to that of a naturally occurring ore of mercury which contains 2.6% mercury. Within another century, the radiotoxicity of irradiated nuclear fuel becomes similar to that of naturally occurring ores of lead (of 5.8% concentration) and of uranium (3% concentration).



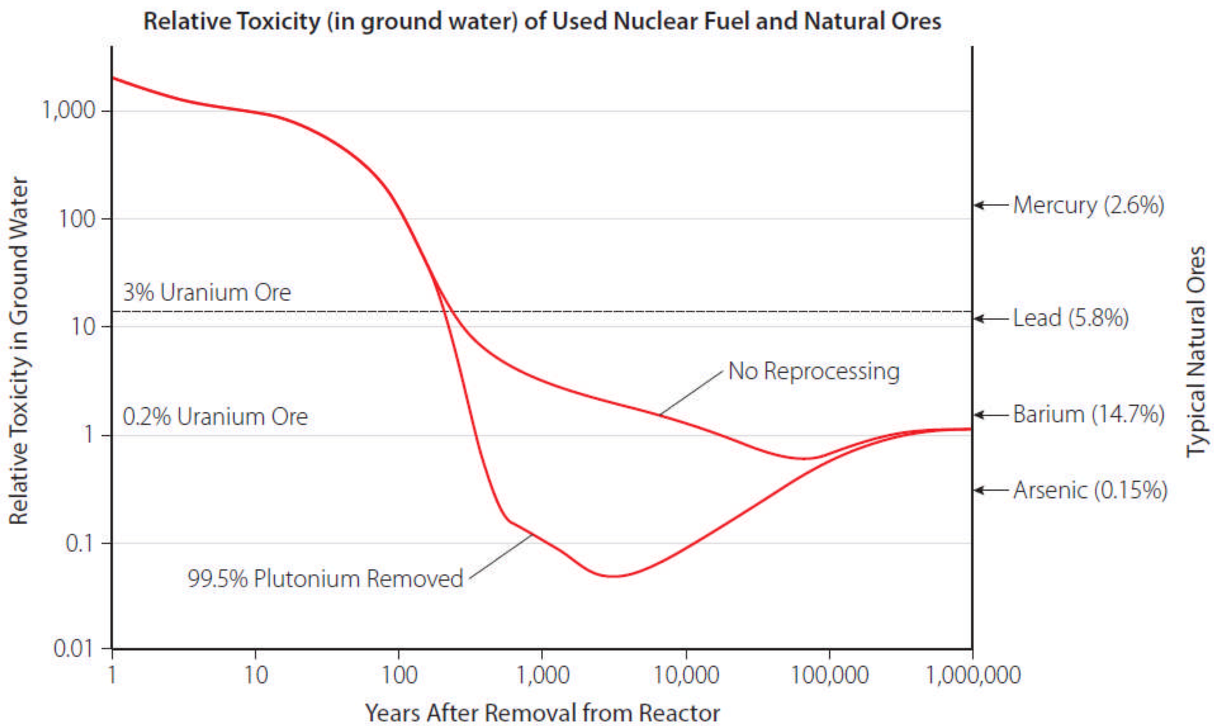
[Illustrative example; Fuel burnup: 220 MWh/kgU]

[Source: Villagran, 2014]

Figure 10 Decay of radioactivity in CANDU fuel

In a repository, access of groundwater to radionuclides is delayed by the engineered barriers described earlier. It must, however, be recognized that some engineered barriers may eventually fail, and therefore it is of key importance to ensure that significant time has elapsed by the time groundwater gains access to radionuclides and carries them to the biosphere. If so, the radionuclides would have decayed to harmless levels by the time they reach the biosphere (as illustrated in Figures 10 and 11). Therefore two important aspects of a safe disposal facility are orderly failure of barriers and the specific sequence of events. Hence, performance modelling is a critical component of the safety case and is explicitly dealt with in the CNSC Regulatory Guide G320 [CNSC, 2006].

Discussions on safety can be conveniently classified into two broad categories: (i) safety during the phase of waste emplacement in the facility, and (ii) safety during the post-closure and entombment phase.



[Illustrative example; Source: Boulton, 1978]

Figure 11 Toxicity in ground water

8.7.2 Operational Phase

The relatively more important aspects of worker safety during this period are:

- radiation dose,
- rock falls,
- container stability during transport,
- hazards and difficulties in retrieval, and
- accident analysis.

8.7.3 Post-closure phase

In any disposal system, potential release of pathogens to the public is the main health concern.

Two important health protection objectives are that the committed dose be negligible and that the risk of increased cancer be negligible. Therefore, an important element of public health protection is provided by making the fuel inaccessible. This is enhanced by entombment of the fuel.

In general, the relatively more important aspects of a disposal facility are container lifetime, radionuclide diffusion, groundwater chemistry, vault sealing, and performance modelling.

It must also be ascertained that the Earth provides sufficient radiological shielding to humans on the surface. Figure 12 shows the results of illustrative dose rate calculations at the surface of a conceptual repository at a hypothetical site. The predicted dose rates provide good margins of safety compared to the natural background dose rate and also compared to the acceptance criteria recommended by the International Commission on Radiological Protection (ICRP).

8.8 Breach of Containment

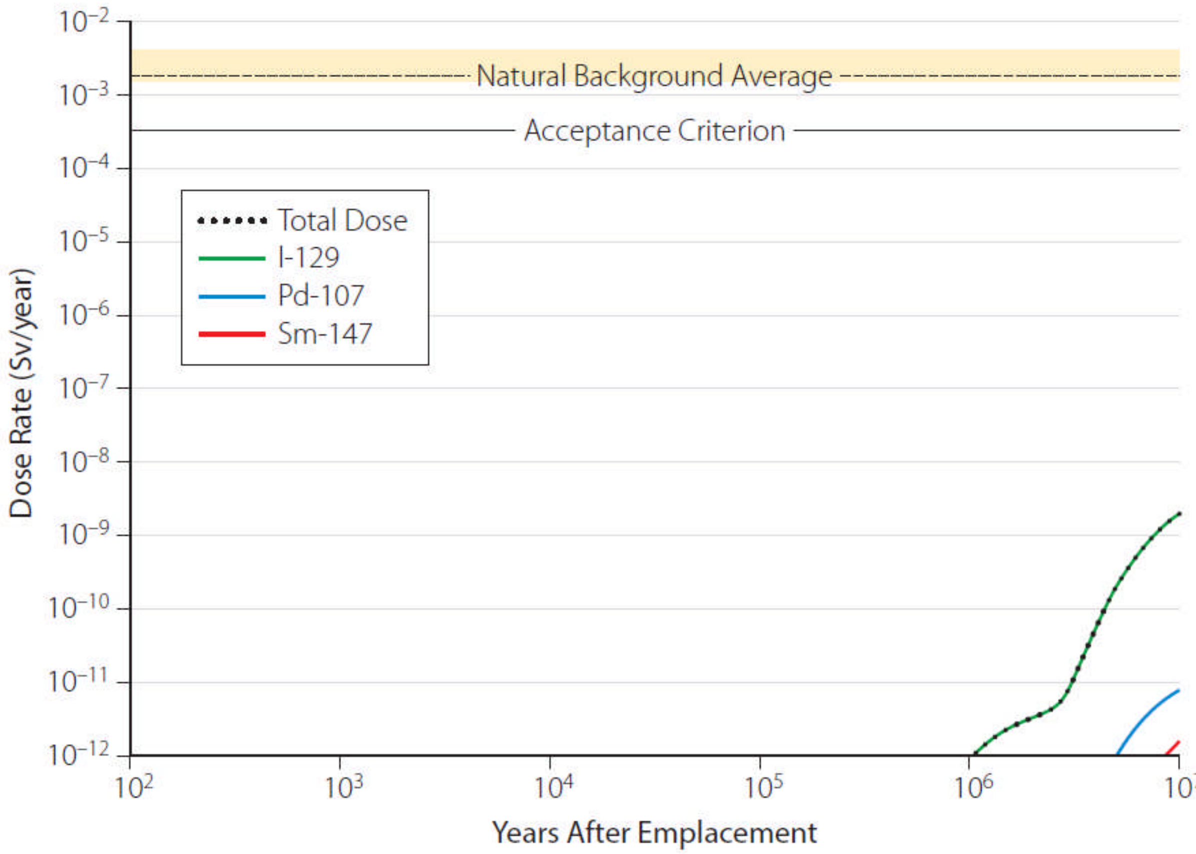
Write-ups in this and the following sections are based heavily on a variety of previous publications, mainly Whitlock [2015].

"In the design of Canadian technology, one aspect that was seriously considered was: 'what if both the copper container and the Zircaloy sheaths are breached?' Would the radioisotopes in used fuel reach the biosphere and pose a significant hazard to human health?"

If both fuel containers were breached, migration through groundwater is the one credible mechanism through which harmful radionuclides from the used fuel could be transported to the food chain and/or to water reservoirs accessible to humans.

Extensive experiments and detailed computer simulations have established that in the current design, transport times to the surface are in the hundreds of thousands of years. In that time, the radionuclides would decay to harmless levels (see Figure 10), and therefore the effects of used fuel on the biosphere would be negligible. In fact, after a few hundred years of decay, the actinide content in the used fuel repository is comparable to that of naturally occurring high-grade uranium deposits found in Canada, as well as those of other toxic ores such as lead and mercury. This period—a few hundred years—is well within human experience for safe design of engineered structures.

Although this science is based on extensive experiments and detailed computer modelling, its complete verification over the timescales of interest is obviously not possible through laboratory experiments alone. Therefore, this technological solution has also been verified using "natural analogues". Natural analogues are systems that occur naturally and possess key attributes that are very similar to components of the used fuel disposal system. They are described next.



[Illustrative calculations; Source: Villagran, 2014]

Figure 12 Dose rate at the surface of a repository

8.9 Nature's Analogues for a Waste Repository

From the discussion above, it should be clear that the two aspects of the Canadian disposal strategy that are most critical to its success and that would benefit from having natural analogues to demonstrate their viability are a demonstration that radionuclide migration through groundwater is equal to or less than that used in the assessment models, and a demonstration that the effectiveness of the buffer material (bentonite clay) is as good as or better than that assumed in the assessment models.

- **Migration of plutonium and uranium**

In 1972, a French uranium mining company deduced that a natural "chain reaction" existed on earth some 2 billion years ago, long before the first chain reaction was demonstrated by human beings. These "natural reactors" existed at Oklo in Gabon, Africa. The key ingredient that made these natural reactors possible at that time is that natural uranium contained a significantly higher proportion of ^{235}U than it does today. At this higher enrichment level, natural uranium, with access to ordinary water, could be configured to constitute a critical mass. In fact, about 16 such "reactor" locales were identified at Oklo. It is postulated that these natural reactors "operated" intermittently for nearly a million years, with their power being regulated by access to water. When the reactor power increased, groundwater heated up and evaporated; this caused the reactor to shut down. When the water cooled down and flooded the uranium deposit, the power started up again, and so on. These reactors shut down permanently when the concentration of ^{235}U became depleted to a value that could no longer sustain a chain reaction. However, the groundwater that had made the chain reaction possible continued to flow over the ore body until it was discovered 2 billion years later.

Unfortunately, by the time the phenomenon of "natural reactors" was understood, most of the locales where these natural reactors operated had been mined and the evidence of radionuclide migration 2 billion years earlier destroyed. Fortunately, some sites were undisturbed, which provided evidence that neither the plutonium (now totally decayed) nor the uranium had migrated from the location in the grain where they existed when fission occurred. This is powerful evidence that the model used for radionuclide migration in the Canadian waste disposal assessments (which assume that the radionuclides percolate to the surface of the repository together with groundwater) is extremely conservative.

- **Effectiveness of the buffer material**

The most direct natural analogue that provides evidence of the effectiveness of the buffer material is the uranium deposit in the Athabasca Basin of northern Saskatchewan, known as Cigar Lake. This deposit was discovered in 1981 and, after a lengthy period fraught with production difficulties unrelated to its geology, came into production in 2014. It is one of the richest (average ore grade of 20%) and largest deposits of uranium in the world. It occurs at a depth of about 500 metres as a contiguous deposit about 2000 m long, 100 m wide, and 20 m thick. A well-defined ridge of basement rocks about 2 billion years old underlies the mineralized zone over its entire length.

The most remarkable feature that has relevance to the effectiveness of the buffer material is the clay deposit that completely covers the ore deposit, on top of which there is an overburden of sandstone. The fact that the ore deposit has survived undisturbed for about 2 billion years speaks to the effectiveness of the clay barrier in preventing water from reaching the ore deposit and leaching out uranium atoms. The effectiveness of this entombment is further verified by the fact that the ore deposit provides no chemical or radioactive signal of its existence that can be detected at the surface. The age of the ore deposit and the traumatic geological events that have occurred during that time, and which events the clay barrier has survived intact, provide indisputable evidence for the effectiveness of the clay barrier. Finally, it must be remembered that the buffer material designated for use in the Canadian disposal facility will be superior to the Cigar Lake clay, and that the host rock in the disposal facility will be far less permeable than the sandstone of the Cigar Lake deposit. Additional thoughts on natural analogues can be found in Whitlock [2015].

9 Problems

Section 2

Q2.1 – Why is it necessary to regulate and control storage and disposal of nuclear waste?

Q2.2 – Why do regulatory bodies such as the IAEA classify nuclear wastes?

Q2.3 – Which Canadian agency is responsible for regulating the manner in which nuclear wastes are to be handled in Canada, and how many classes of nuclear waste are designated?

Section 3

Q3.1 – What are the two stages of nuclear fuel storage that are generally accepted and used in Canada?

Q3.2 – Why is nuclear fuel stored in water-cooled pools immediately after discharge from the reactor and for several years thereafter?

Q3.3 – After being stored in a water-cooled pool for several years, why is it acceptable and desirable to remove irradiated CANDU fuel bundles from the pool and store them in air-cooled storage cylinders?

Section 4

Q4.1 - Designers of storage facilities must be mindful of a number of design requirements that must be met to ensure the integrity and security of irradiated fuel while it is being stored. List them.

Section 5

Q5.1 – What are the two overriding requirements specified by the IAEA that must be met to ensure the integrity of spent fuel while it is being stored?

Q5.2 - Whether stored in water-cooled pools or air-cooled storage cylinders, irradiated fuel and its surroundings are potentially susceptible to a number of damage mechanisms and other challenges. List them.

Section 6

Q6.1 – What is the preferred orientation for CANDU fuel bundles in-reactor and in storage facilities?

Q6.2 – Why are irradiated CANDU fuel bundles required to reside temporarily in air as they pass from the fuelling machine to the irradiated fuel discharge room?

Q6.3 – How are the irradiated bundles, when residing in air, prevented from overheating?

Q6.4 – How much water is contained in a typical irradiated fuel storage pool?

Q6.5 – What can the operators of irradiated storage pools do to minimize oxidation of fuel Zircaloy components?

Q6.6 – How are the workers inside irradiated fuel storage facilities protected from radiation produced by irradiated bundles?

Q6.7 – What is the optimum stacking arrangement of irradiated CANDU bundles to prevent criticality in the irradiated fuel bay?

Section 7

Q7.1 – Describe the main features of a storage cylinder.

Q7.2 – Although the thermal power of the irradiated bundles decreases dramatically while the bundles are stored in water-cooled pools, the bundles' emission of ionizing radiation remains high (due to decay of long-lived isotopes and elements). How are people and the environment protected from radiation when bundles are stored in concrete casks?

Section 8

Q8.1 – What types of geological formations are thought to be suitable for final entombment of irradiated fuel (or fuel waste)?

Q8.2 – What is the proposed method in Canada for permanent disposal of high-level radioactive waste?

Q8.3 – What is "adaptive phased management" and how does it influence Canada's high-level waste disposal program?

Q8.4 – What are the two distinct phases in the "life" of a disposal facility?

Q8.5 – Describe the main components of a disposal facility.

Q8.6 – What are the two physical properties of bentonite clay that make it eminently suitable for sealing storage containers inside rock caverns?

Q8.7 – Once the irradiated bundles have been placed into containers and the containers sealed inside the vault, how many barriers are at work to prevent the radioactivity from affecting people and the environment? Name the barriers.

Q8.8 – What are the life-limiting factors for a storage container, and what is the container's expected lifetime?

Q8.9 – What is the principal characteristic of the Deep Geological Repository that provides shielding protection to people and the environment from the radiological effects of irradiated fuel?

Q8.10 – The initial three barriers to release of radiologically harmful effects of irradiated fuel to people and the environment are UO_2 pellets, Zircaloy sheaths, and steel/copper containers, which can delay the release of radioactivity for several million years and are assumed (conservatively) to do so for 10^5 years. If these three barriers were accidentally breached immediately after the fuel was entombed, how would the effects of radiation on people and the environment be affected?

Q8.11 – Empirical verification of the efficacy of the proposed disposed concept would require several hundred years, if not hundreds of thousands of years, which is not feasible. Therefore, verification to date has been based on short-term experiments, computer modelling, and natural analogues. Name and describe the natural analogues that have produced information that provides direct support (verification) of the design of the disposal facility proposed in Canada.

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12 Relationships with Other Chapters

Chapter 8 provides an overview of fuel bundle configuration. Chapters 3 to 5 explain the neutron physics that generates heat in the fuel. Chapters 6 and 7 explain how that heat is removed from the fuel bundle and illustrate the internal temperature distribution within a fuel rod. Chapter 13 explains fuel performance during postulated accidents. Chapters 14 and 15 explain chemical and metallurgical aspects that relate to fuel sheath corrosion. Chapter 17 describes fuel design and performance, focussing on the current natural-uranium cycle (i.e., low burnup). Chapter 18 describes a few selected fuel cycles, summarizes key aspects of fuel manufacturing related primarily to the natural-uranium fuel cycle, and explains a few selected aspects of fuel performance that become relatively more important for advanced fuel cycles. Finally, the current chapter (#19) describes interim storage and disposal of irradiated fuel.

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