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A review of nuclear spent fuel storage facilities

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ABSTRACT

The objective of this review paper is to summarize the current state of understanding on a topic 'Nuclear spent fuel storage and facilities'. This review paper surveys and summarizes previously published studies, rather than reporting new facts or analysis. It is importance to case study this issue since the number of spent fuel are increasing in which a typical nuclear power plant in a year generates almost 20 metric tons of used nuclear fuel. In which the nuclear industry generates a total of about 2,000 - 2,300 metric tons of used fuel per year and for the last 40 years produced 76,430 metric tons of nuclear spent fuel. Future understanding and attention need to be accomplished since spent fuel can cause harm due to its high radioactive level and also the ability to reprocess the fuel to be used as MOX fuel.

1. Introduction

Nuclear reactor all around the world produces waste which contain harmful radiation to the environment and the human. The waste comes from the fuel which no longer have sufficient composition of Uranium-235 to maintain fission reaction. Every 1 or 1 and half years, the fuel will be changed with a new fresh fuel. The fuel that has been removed out is called spent fuel and is stored inside the spent fuel pool.

Conceptually, storage implementation was meant to act as an intermediate plan before a permanent decision is made. Hence, the ultimate goals of spent fuel dry storage are to prevent a gross rupture of the spent fuel during operation and to keep its retrievability until transportation [1]. These objectives in turn lead to active spent fuel integrity evaluation research, which is important

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before storage system development. For example, the main safety evaluation checklists of dry storage system are structure, confinement, shielding, criticality, thermal, and material degradation, while the last two items are directly related to spent fuel integrity. Specifically, peak rod temperature in the dry storage system is a significant spent fuel integrity evaluation item, which has a great influence on the system design criteria. It is the only controllable parameter in a passive cooling system in dry storage [2].

Virtually all power reactors have some form of spent fuel pools associated with the reactor operations. In recent years, at-reactor spent fuel storage has been used to full capacity in some cases, threatening the continued operation of the power plant. Recent designs of the reactor have in fact now incorporated pools that can accommodate lifetime arising over periods of up to 40 years. However, older operational plants, due to their limited at-reactor capacity have necessitated the development of away-from –reactor. Two technologies have been developed for storage. Initially the storage method was wet, but in recent decades, dry storage techniques of varying types have been developed. Away-from-reactor storage can be considered in two categories. The first is where additional interim storage capacity is constructed at the reactor site but largely or entirely independent of the reactor and it is at-reactor pool The second category of away-from reactor-storage is off reactor site at an independent location. A large portion of this away-from-reactor is in form of pools are processing plants particularly in France, United Kingdom and The Russian Federation [3].

There are two acceptable storage methods for spent fuel which are spent fuel pools and dry cask storage [4]. The spent fuels will be cooled in the spent fuel pools for at least one year and often as much as ten years before sending them to dry cask storage. Dry cask storage is a method of storing high-level radioactive waste, such as spent nuclear fuel [5].

2. Spent fuel and material facilities

2.1. Spent nuclear fuel

Spent nuclear fuel is the term used for nuclear fuel which no longer effective for fission process in order to produce sufficient heat inside of the core of nuclear power plant. However, spent nuclear fuel still produces some heat, and high in radioactivity level [6]. It is believed that spent fuel still could be harmful towards its surrounding, causes it becomes a serious concern due to its contents (actinides, and fission products) [7]. Thus, proper management of spent fuel is strongly proposed.

The properties of spent fuel are basically different from the fuel properties during operation, especially in temperature aspect. During operation, the temperature of fuel is approximately 1000^{o}C, whereas the temperature of the spent fuel is believed to be equivalent to the possible temperature measured at the surface of cladding of the fuel when it is stored in dry storage, which is in range of 300^{o}C until 400^{o}C. The typical end-of-life (EOL) pressure at room temperature (environment temperature, not the spent fuel temperature) is approximately 1.5 MPa, and in the range of 3 to 4 MPa (for LWR) [8].

From the Fig. 1 above, basically, the major element present in the spent fuel is non-fissile uranium which is Uranium-238. This is amount of U-238 left, which is 95% after power generation, from 97% at initial. The remaining 2% loss from the initial content contributed into Plutonium and Uranium-235 production [9]. The fission products found in the spent fuel composition are the products of the reaction between U-235 and neutrons (fission process) which contributed in energy production [10]. From this comparison, it can be assumed that the majority of the spent fuel compositions are not classified into waste, but it can be treated as reusable material [11].

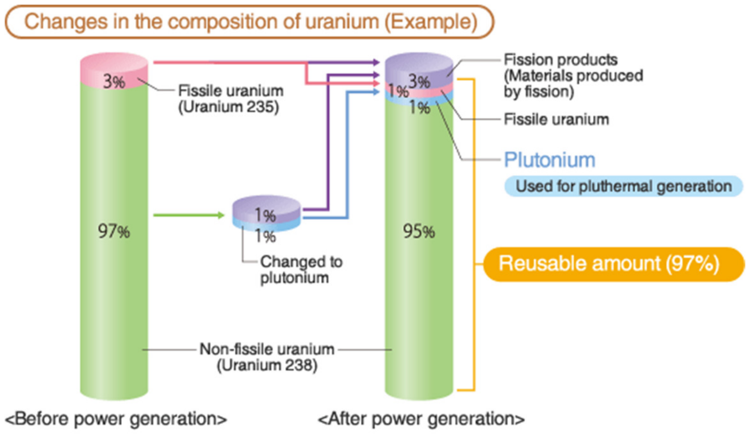

Fig. 1. The changes of the composition of spent nuclear fuel [5]

Spent nuclear fuel from power reactors is unloaded into a water-filled pool immediately adjacent to the reactor to allow its heat and radiation levels to decrease [12]. The composition, heat output and radioactivity per ton of heavy metal of the spent fuel depend upon the burn-up. Spent fuel consist of uranium, plutonium, fission product, minor transuranic elements and actinide [13-15].

2.2. Spent fuel integrity evaluation

Nuclear fuel degradation directly affecting integrity of spent fuel while fuel temperature is the main reason for fuel degradation. In a dry storage condition, the effect of fuel temperature can be more dominant because other mechanisms except creep and hydrogen effect had already stopped its progression after reactor operation. Fig. 1 shows the relationships among the various nuclear fuel degradation mechanisms, which are established and caused based on the fuel temperature [2]. On the other hand, the cladding material condition for nuclear fuel is dependent on the reactor's operation history and also the storage temperature history. Again, fuel temperature and burnup can play a crucial role in determining the thickness of oxide layer, fraction of hydrogen pickup, rate of releasing fission gas, pressure of rod internal, and hoop stress on the inner cladding wall. With the establishment of spent fuel evaluations, we can set the safety criteria for fuel temperature which to be lowered and this can be done through calculating the temperature of all system components. Thus, accurate estimation of the actual fuel cladding temperature and evaluation of remaining engineering margin of the cladding within a target dry storage operation lifetime is very essential [17].

2.2.1. Material degradation

A lot of spent fuel placed in wet storage of plant sites might have been exposed several times to thermal cycling during reactor operation and replacement of the storage rack or transfer to a neighbouring storage facility to increase some on-site storage capacity [17]. Spent fuel stored in the wet storage would be moved to a dry storage facility. This significant change in storage condition can cause fuel degradation against the long-term of the spent fuel. Fuel degradation can occur as a result of mechanisms that may affect the cladding integrity of spent fuels during dry storage, and subsequent handling and transportation operation are air oxidation, thermal creep, stress corrosion cracking (SCC), delayed hydride cracking (DHC), hydride re-orientation, and hydrogen migration and re-distribution [18,19]. The degradation mechanisms usually related to the safety issues for fuel transfer to the disposal facility.

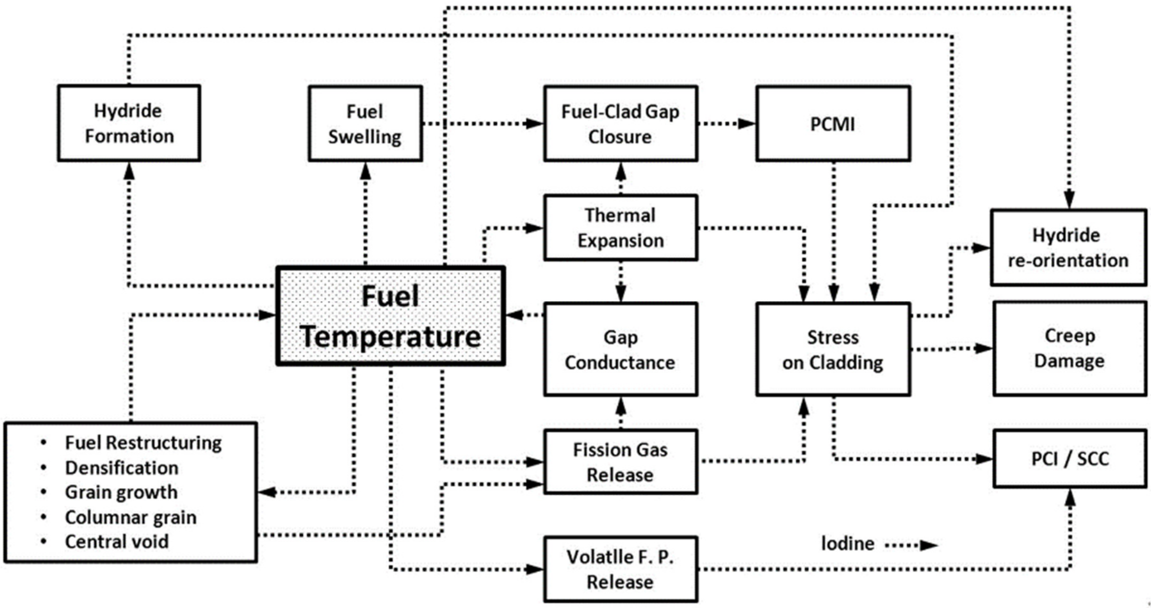

Fig. 2. Nuclear Fuel Rod Degradation Relationship [2]

Hydride reorientation can occur when part of the hydrogen created by oxidation of the cladding by water is absorbed by zircalloy during reactor operation, and once the solubility limit of hydrogen in the cladding material matrix is reached, hydrogen precipitates is formed [20]. The important parameters include material type, hydrogen content and distribution, fluence, temperature, applied stress, cooling rate and extent of temperature variations when thermal cycling is involved. These parameters are related with the design parameters of the storage and transfer system which can be implemented in the safe storage conditions.

Stress corrosion cracking (SCC) is a situation where a complex phenomenon driven by the synergistic interaction of mechanical, electrochemical and metallurgical factors. SCC is the term given to crack initiation and sub-critical crack growth of susceptible alloys under the influence of tensile stress and corrosive environment. Oxidizing conditions would lead to increased formation and transport of corrosion products, higher radiation fields, crud buildup on fuel, increased corrosion of fuel rods, and increased susceptibility of some structural materials to SCC. Minimization of coolant oxygen concentrations will lead to minimization of both SCC and general corrosion in the Reactor Coolant System. Dissolved oxygen concentrations can be controlled during plant heat-up by venting or vacuum filling followed by the use of hydrazine or hydrogen for residual oxygen scavenging.

Oxidation occurred in the spent fuel due to the interaction between water and zircaloy as a control rod. Oxidation can lead to radionuclide release from spent fuel under dry conditions. Dry-air oxidation played an important role in radionuclide releases. UO2 matrices will crack upon oxidation by volume change. Lower oxides such as UO2 will contract, whereas higher oxides such as U3O8 or UO3 hydrates will expand. Lower and higher oxides are defined as oxides of (O/U) ratio smaller and larger than 2.4, respectively. Fracture can increase the exposed surface area of a spent fuel surface to groundwater or to air. This process can lead to the releasing of gaseous radionuclides such as C-14, Cl-36 and I-129 and high-solubility radionuclides such as C-14, I-129, Tc-99 and Cs-135 [22].

Fig. 3. Stress Cracking Corrosion [21]

Thermal creep is a time dependent deformation under an applied load. It consists of 3 stages which are primary, secondary and tertiary as in Fig. 3. In primary stage, the creep is in initial rapid and slows with time. Secondary stage shows that it is in uniform rate whereas in tertiary stage, the creep is at accelerated rate which leads to rupture. Thermal creep usually only occurs at high temperature. When the creep is greater, then the thinner the fuel clad wall becomes and cause a greater potential for the fuel clad to rupture.

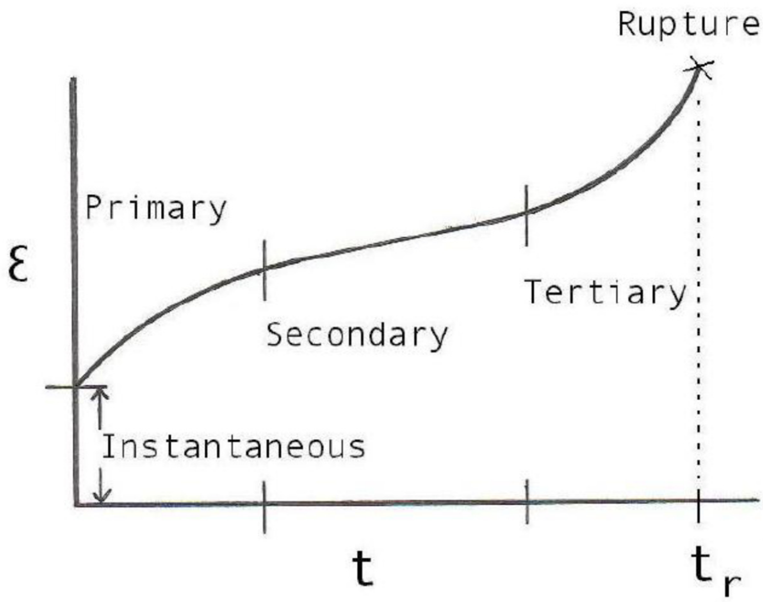

Fig. 4. Creep stages [23]

Delayed hydride cracking (DHC) is a sub-critical crack growth mechanism occurring in zirconium alloys that requires the formation of brittle hydride phases at the tip of a crack and subsequent failure of that hydride resulting in crack extension. Hydrogen in solution is precipitates as a hydride phase when it is transported to the crack tip by diffusion processes. The crack tip hydride grows due to the migration of hydrogen in the bulk of the material at some characteristic distance from the crack tip.

The driving force for the diffusion of the hydrogen is the difference in the chemical potential of hydrogen in the crack tip hydride due to local hydrostatic stress at the characteristic distance from the crack tip [24].

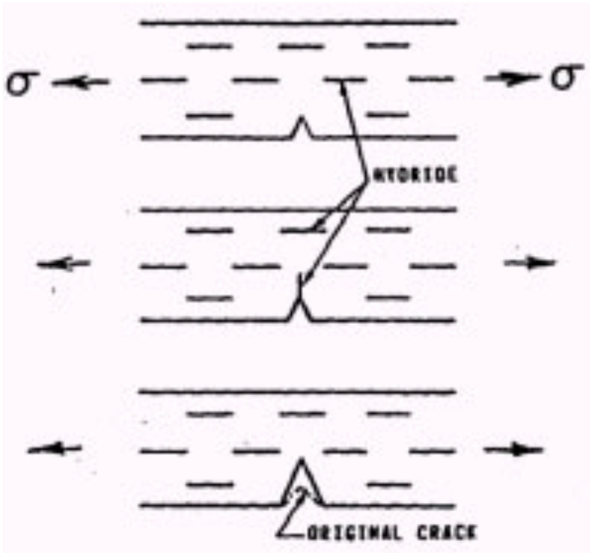

Fig. 5. schematic illustration of a single step in delayed hydride cracking [24]

2.2.2. Status of spent fuel integrity evaluation in advanced countries

Since in the middle of the 1970's, some advanced countries such as Germany and USA already have executed the spent fuel integrity evaluation just after cancelling the reprocess policy for spent fuel management. Based on their 10 years of research outcomes, they established almost the same peak rod temperature criteria methodologies for a dry storage system. Started from 1985, they applied these principles to a commercial dry storage system [25]. Japan, as one of the advanced Asia country, has also performed spent fuel integrity evaluation for almost 10 years to establish its own criteria methodology before building the massive Away-From-Reactor (AFR) dry storage in Mutsu city [2].

2.2.2.1. Status of USA

During the 1970's to 1980's, Commercial Spent Fuel Management (CSFM) [26] program is the authority who lead the spent fuel integrity research in USA. As a final outcome, the most important degradation mechanism for cladding integrity in dry storage was creep by referring to creep and creep rupture maps [27]. To stop progression of creep, they have found that spent fuel species, cladding materials, burnup levels, and charging gases in the storage system are affecting the independent storage temperatures. These various temperature limits for each system were applied until 1997 [28].

In 1999, the published Interim Staff Guide (ISG)-11 excluded Diffusion-controlled Cavity Growth (DCCG) as an important cladding degradation mechanism even the initial NUREG-1536 report which published in 1997 included it [29]. Excluding DCCG is due to its extremely low occurring possibility in dry storage operation and used 'low burnup spent fuel' to represent lower than 45,000 MWd/MtU burnup spent fuels. In 2000, ISG-11 has been revised (Rev.1) and it considered the oxide layer thickness and restricted less than 1% creep of spent fuel cladding [30]. In 2002, the secondary revision, ISG-11 (Rev.2) found out that creep is not considered as a critical threat to spent fuel integrity because it shows a self-limiting phenomenon even it is the most important degradation mechanism of spent fuel cladding in dry storage. In 2003, the third revision of ISG-11 (Rev.3) [31] proved that the maximum number of thermal cycling as less than 10 cycles according to technical research work by Kammenzind [32].

Table 1 listed the most considerable degradation mechanisms for fuel and cladding for additional storage, estimated for 40 years from 2005, which suggested by the technical bases report [33]. As we can see from the table, various considerable degradation mechanisms for additional storage was reduced, and that it concentrates on off-normal and accident conditions. Furthermore, the hydride effect is becoming significant as the storage system temperature drops. Also, from this table, we can expect which degradation mechanism can be dominant in each time frame using the temperature range for that time, and realize that the fuel pellet has no effect and creep is no longer critical in additional storage, and recognize that because of transportation at the end of storage, cladding degradation induced by hydride is increasing in importance as time goes by and the system temperature drops.

Table 1
Considerable Degradation Mechanism for Additional Storage [33].

Mechanism	**Fuel**		**Cladding**	
	Initial	Extended	Initial	Extended
Normal				
Oxidation	N	N		
FGR	Y	N		
Creep			Y	N
DCCG			N	N
H_2 reorientation			Y	N
DHC			N	N
SCC			May be	N
H_2 embrittlement			N	Y
H_2 migration			Y	N
Annealing			Y	N
Crud Spallation			May be	N
Off-Normal				
Oxidation due to air ingress	Y	Y	Y	Y
Creep			Y	DE
Annealing			Y	DE
Accident-impact				
Fracture	DE	DE		
Oxidation	DE	DE		
Impact Breach			DE	DE
Crud spallation			DE	DE
Accident-fire				
Stress rupture			Y	Y
Annealing			Y	Y
H_2 reorientation			Y	Y

DHC=Delayed Hydrogen Cracking; DCCG=Diffusion Controlled Cavitation Growth; FGR=Fission Gas Release; SCC=Stress Corrosion Cracking; DE=Depends on the Event

NUREG-1536 Revision 1 was published in 2010, which concluded all previous technical changes and categorized the operation mode as normal, off-normal, and accident. The definition of an off normal condition means ‘once per year’, and an accident condition typically means ‘once per system lifetime’ from a probabilistic point of view. This revision has also deployed the risk assessment concept to regulator's reviewing process by classifying all check points as high, medium, or low level. This risk assessment concept for every dry storage 'system, structure and components' (SSC) has been popularly applied to an 'extended storage gap analysis', which is an issue worth discussed in the USA.

2.2.2.2. Status of Japan

In Japan, a new domestic reprocessing facility, the Rokkasho Reprocessing Plant (RRP) has been delayed starting its commercial operation due to a technical problem, according to reprocessing policy for spent fuel in their own country and because of few consignments of spent fuel reprocessing in other countries have been accomplished until 2003 [34]. The spent fuel accumulation problem in Japan is becoming critical when RRP failed to handle capacity of 800tU per year in 2010. This scenario brought up the issue of necessity of massive dry storage before a working reprocessing facility starts to operate [35]. Fig. 6 shows the spent fuel accumulation projection and its expected solutions in Japan.

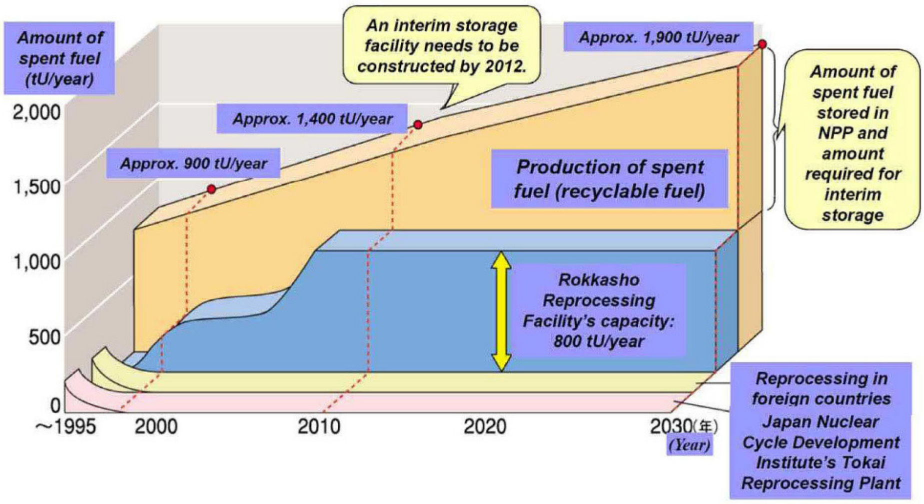

Fig. 6. Spent Fuel Accumulation Projection and its Expected Solutions [35]

The final goal of spent fuel storage in Japan is to keep its integrity by prevent thermal creep failure and stopping loss of mechanical properties due to cladding degradation. After a 20 years of spent fuel integrity research for only one rod has been carried out at Central Research Institute of Electric Power Industry (CRIEPI), it was concluded that there was no significant problem with this test rod [36]. Based on the above findings, a small quantity of dry storage can be installed at the Fukushima and Tokai sites. But, a total different concept is needed for a new largescale necessity of dry storage. In Japan, the core suggest of dry storage is a 'Holistic Approach', which means closing of a cask lid is necessary until the reprocessing step using a Dual Purpose Dry Metal Cask (DPDMC) for 50 years of storage and transportation to the reprocessing facility [37]. Therefore, there's a need for their own technical criteria for dry storage. For spent fuel integrity criteria, the Japan Nuclear Energy Safety (JNES) supported by the Nuclear and Industrial Safety Agency (NISA) has studied thermal creep, hydrogen distribution & hydride reorientation, and irradiation hardening & recovery using various burnup range spent fuels for about 10 years [34].

2.3. Spent fuel storage facility

Generally, there are three main types of nuclear power plant being used worldwide today. They are light water reactor (LWR), heavy water reactor (HWR) and advanced gas reactor (AGR). LWR can be divided into pressurized water reactor (PWR) and boiling water reactor (BWR). These reactors used light water which is commonly known as 'common water (H_2O) as coolant and moderator. The difference is PWR has two uncontaminated loops and has pressurizer to maintain the water from boiling. While BWR has only one contaminated loop and has no pressurizer. Hence, the water is boiling to transfer the heat to the steam turbine.

HWR is the second most common type of reactors in used. Pressurized Heavy Water Reactor (PHWR), also called Canada Natural Uranium Deuterium (CANDU), used heavy water D_2O as the coolant and moderator. The third most common is AGS which used carbon dioxide (CO_2) as coolant and graphite as moderator.

An average modern reactor has a capacity of about 1 GWe (1 gigawatt-electric or 1,000 megawatts-electric). According to the International Atomic Energy Agency, there exist today 331 GWe of LWRs, 23 GWe of PHWRs, and 19 GWe of graphite-moderated reactors. Almost all the reactors now under construction are LWRs, and indeed most are PWRs [38,39].

The amount of spent fuel discharged from a nuclear power plant depends upon the fuel "burn-up," i.e., the thermal energy (heat) generated per unit mass of fuel [2]. Table 1.1 shows the approximate amount of spent fuel that would be discharged per year from a 1 GWe reactor of the three most common reactor types [40].

Table 2
Annual discharge of spent fuel for three common reactor types. This assumes a reactor of 1GWe operating at 90% capacity. GWd/tHM is the amount of thermal energy (heat) in gigawatt-days released per metric ton of heavy metal (HM) in the fuel.

Reactor type	Typical burn-up (GW d/tHM)	Annual discharge of spent fuel (tons)
LWR (light-water moderated)	50	20
CANDU)heavy-water moderated)	7	140
RBMK (graphite moderated)	15	65

2.3.1. Swedish containers for disposal of spent nuclear fuel.

The purpose of a storage facility is to isolate the radioactive waste from man and environment. If the isolation is broken, the leakage and transport of radioactive substance must be retarded [41]. The package is one of several barriers, used to achieve these two main functions. For short-lived, low and intermediate level waste, four standard containers of steel and concrete are used. Spent fuel will be placed in a canister consisting of pressure-bearing insert of cast nodular iron and an outer corrosion barrier of copper before it is deposited in a deep geological repository. In particular, the development of high integrity copper canister for isolation of spent fuel is described in this paper [42]. In Spent Fuel Respiratory (SFR), the FINAL Respiratory for Operational Radioactive Waste, in operation since 1988, four standard containers are used for storing short-lived, low and intermediate level waste. The containers are as shown in Fig. 1 [43].

Cubic concrete mould with sides of 1.2 m where the thickness of the wall is between 10 to 35 cm, are one type of container used. Moulds in the same size are also constructed in steel. The material is carbon steel that has been painted with epoxy to prevent corrosion. Solidified waste could also be placed in seal drums with a diameter of 0.6 m and height of 0.9 m. the drums can be placed on plates or in boxes four by four, for better handling inside the SFR facility. The tanks have a base area of 1.3

m × 3.3 m and height of 2.3 m. Besides these four specially built containers, standard ISO-containers are used for solid low-level waste, which is packed in primary package. The containers are produced with a strong steel frame covered with corrugated sheet iron and are available in different sizes [41].

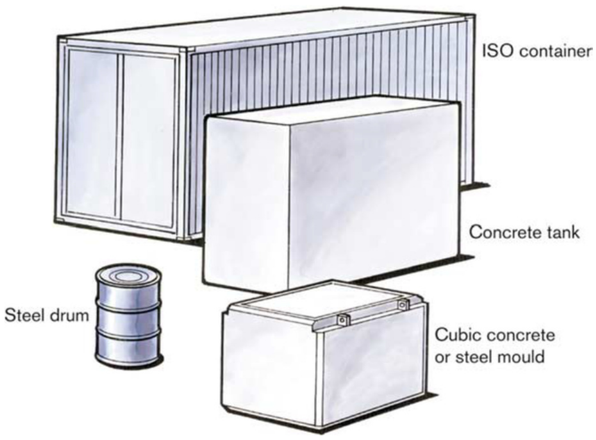

Fig. 7. Standard containers for spent fuel respiratory

The spent fuel was encapsulated with a canister before it is placed in a deep respiratory. The canister was designed to consist of an outer corrosion barrier of copper and pressure-bearing insert of cast nodular iron. The canister was illustrated in Fig. 2 [41]

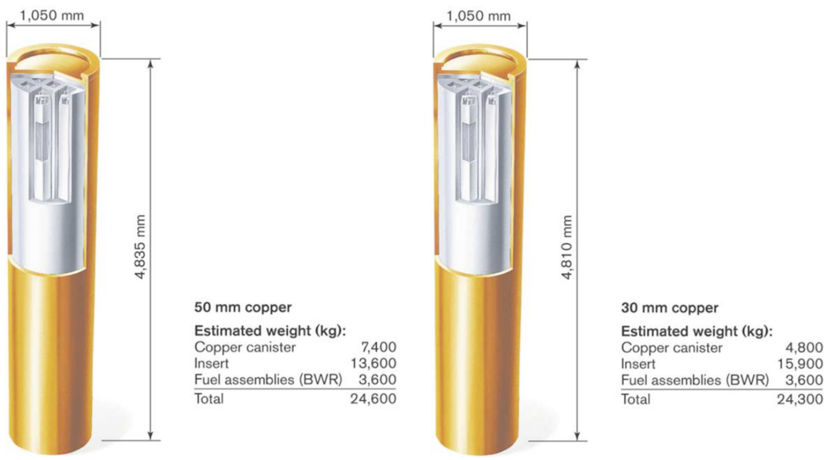

Fig. 8. Planned design of the Swedish spent fuel canister [3]

2.3.2. Concrete silo dry storage for CANDU spent fuel

For fuel generated from CANDU reactor, concrete silo dry storage system is used. The silo is designed to remove passively the decay heat from the spent fuel as well as the integrity of spent fuel during storage period. Dominant heat transfer mechanism must be characterized and validated for the thermal analysis model of the silo, and the temperature history along storage period could be determined by using the validated thermal analysis model [44].

The spent fuel storage is a cylindrical silo made of stainless steel type 304L with external diameter of 1.06m and height of 0.56 m, and capable of storing sixty bundle of CANDU spent fuel. The basket consists of two subassemblies that are the basket base and the basket cover. The basket base provides both support for the fuel and the means to handle the basket. It consists of horizontal base plate, a vertical lifting post welded to the center of the base plate and a horizontal positioning plate. The positioning plate has sixty holes to accommodates the fuel bundle, thus provide the individual lateral support at the bundle wear pads. The central post has a diameter of 10.16 cm, schedule 120 pipes to lifting collar has been welded. The basket cover has a horizontal plate. The cover has a hole in the center to accommodate the central post of the base subassembly. [44].

The concrete silo has cylindrical reinforced concrete structure with external diameter of 3074 mm. internal diameter of 1118 mm, height 6518 mm and thickness of 978 mm. the silo was designed to have a carbon-steel liner plates installed on the internal surface, and the steel liner was coated with epoxy paint the prevent corrosion due to ambient air. The silo is capable to piling nine fuel baskets. The top of the silo was opened to allow the entry of the basket. Once the nine baskets were filled inside the silo, the silo was sealed with a silo plug. The silo was design is as shown in Fig. 3 [44].

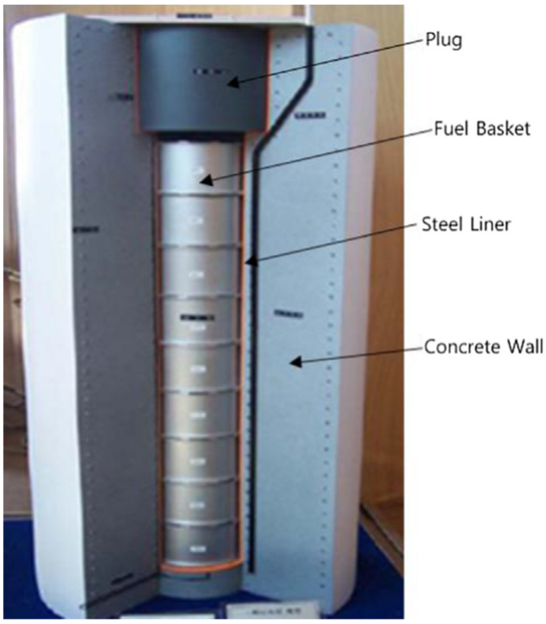

Fig. 9. Concrete silo structure

2.3.3. Waste management plans for ITER

ITER produce five type of nuclear waste. Each type of waste generated from ITER was disposed in different ways. For solid long-live intermediate level waste, the waste mainly remotely removed from the vacuum vessel and transferred by cask to the hot cell building. For solid purely tritiated waste, the waste was characterized and packed in drum and interim stored in the tritium building before being sent to a storage area in the hot cell building. For the solid short-lived, low and intermediate level waste, the waste is transferred to the radwaste building where a several treatment is foreseen. Once cemented and cured, the packaged are stored in the radwaste building for six months before being exported. Solid very-low level waste is transferred in drum to the personnel and access control building where it is characterized and stored in packages for six months before being exported. Liquid radioactive waste is transferred by pipe from the tokamak building the radwaste building, treated and solidified. The solidified concrete then undergoes the same route as the short-lived low and intermediate level waste [45]. The waste was temporary stored in ITER facilities and will sent to the Andra final disposal or to an interim storage for tritium decay if the waste cannot be directly accepted by the Andra because tritium content and outgassing [46].

Table 3
Estimation of the tritium activity in the radwaste produced during operation.

Type of waste	Tritium activity (Bq/g)
VVLW	See note
LILW-SL	See note
ILW-LL [7] (before detritiation)	Up to 1E+09
Purely tritiated waste [7]	Up to 2E+06

Note: to comply with the present disposal acceptance criteria, the limits of the tritium radwaste activity at the beginning of a 50-years decay period correspond to 15E+03 Bq/g (VLLW, IRAS – 1) and 3E+06Bq/g (LILW-SL).

- VLLW: very-low-level-waste: this waste results from normal operation and maintenance activities. Generally, it includes gloves, paper, clothes, etc. which categorize as incinerable and non-incinerable housekeeping waste. There are also coming from decommissioning of ex-vessel components.
- LILW-SL: low-and intermediate-level short-lived waste: similar with VVLW but higher activity.
- ILW-LL: intermediate-level long-lived waste: this waste comes from the maintenance of in-vessel components, like, blanket, divertor, ion and electron cyclotron heating and current drive systems, and pumps). Furthermore, the activity of removing the components from the vessel during the de-activation phase under IO's responsibility also produced this type of waste.
- Purely tritiated waste: the activity of maintenance and decommissioning of the tritium plant and fuelling system produced this type of waste.

No HLW (high-level waste) being produced during the operation and decommissioning of ITER.

2.3.4. Studies around the world
2.3.4.1. Canada

Atomic Energy of Canada Limited (AECL) and several Canadian utilities are Canada's government-owned nuclear R&D and reactor-construction organization has created the Nuclear Waste Management Organization (NWMO). This is to recommend a path forward and to analyse to select suitable site for nuclear reactor waste in 2002. In 2005, NWMO has introduced "Adaptive Phased Management" for disposing waste in deep geological repository with any possible monitoring and retrieving the fuel after 240 years of emplacement. This organization also introduced various criteria for site selection. Currently, all spent fuel in Canada is stored at the reactor sites in pools and dry storage. NWMO does not anticipate commencement of a repository until 2035 [47].

The first reactor in Canada is Zero Energy Experimental Pile at Chalk River, Ontario in 1945 then National Research Experimental (NRX) reactor in 1947. As June 2011, Canada already have 18 power reactors with total generating capacity of 12.5 GWe (net) located at Ontarion, Quebec and New Brunswick. All Canada's NPP are heavy-water reactor which is fuelled by natural uranium.

Until June 30, 2010, Canada had 2.2 million spent fuel bundles in storage including 1.54 million in wet storage and 0.66 million in dry storage. Each bundle contains about 20kg of uranium and 44,000 tons of heavy metals. Currently in Canada, all spent fuel is stored in interim wet or dry storage near the power plant for several years. The period of storage is varying from site to site. After some times, the spent fuel is transferred to interim dry storage also near the power plant since NWMO is not build any repository till year 2035. Canada has used three designs which are: AECL silos, AECL MACSTOR and OPG dry storage containers. AECL silos stand for AECL Concrete Canister Fuel Storage Program was created in the early 1970s to demonstrate that dry storage for irradiated reactor fuel was a feasible alternative to continued water pool storage [48]. These canisters can hold between 324 to 600 bundles of fuel. AECL MACSTOR (Modular Air Cooled Storage) have seven modules and each modules holds 12,000 fuel bundles and contains about 230 tons of uranium in spent fuel. These two types of storage are being used outdoors.

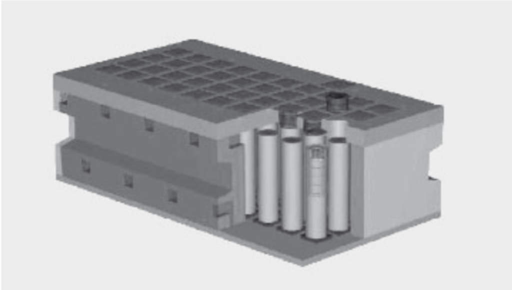

Fig. 10. schematic of the interior of a module. Source: Atomic Energy Canada Limited [47]

Fig. 11. Spent fuel storage modules (MACSTOR) at Gentilly-2 reactor in Quebec [47]

Table 4
Inventory of spent fuel in storage in Canada as of June 30, 2010 [48]

Site	Current net power capacity (GWe)	Number of fuel bundles in wet storage	Number of fuel bundles in dry storage	Authorized dry storage capacity
Bruce A and B Nuclear Generating Station (ON)	4.60	739 947	192 376	750 000
Darlington Nuclear Generating Station (ON)	3.512	329 198	48 363	575 000
Douglas Point Waste Management Facility	-		33 356	
Gentilly-1 Waste Management Facility[33]			3 213	
Gentilly-2 Nuclear Generating Station (QC)	0.635	29 833	86 340	240 000
Pickering A and B Nuclear Generating Station (ON)	3.094	401 737	214 436	633 600
Point Lepreau Nuclear Generating Station(NB)	0.635	40 758	81 000	180 000
McMacter Nuclear Research Reactor (ON)	-	40		
Chalk River Laboratories (ON)	-	367	9 576	
Whiteshell Laboratories	-		2 268	

Ontario Power Generation (OPG) dry storage facilities were an indoor storage and used containers that easy to transport. Each container can hold about 384 bundles and weight almost 60 tons when empty and 70 tons when loaded. The containers were in rectangular shape and the walls were built with concrete inside and the outside were carbon steel, while, the container cavity was flushed with helium gas to protect the fuel bundles from oxidize with the wall where helium gas is inert gas and did not interact with any metal since its already at stable state.

2.3.4.2. France

France has a very systematic system on managing their nuclear spent fuel. They even have introduced acts that related to the management of spent fuel. One of them was "Bataille Act" which introduced in 1991 that required 15 years of R&D including an analysis of schemes to separate and transmute long-lived radioisotopes to shorter-lived radioisotopes. In 2006, another two acts have been introduced which were "Act on Transparency and Safety in the Nuclear Field" and "Act on Sustainable Management of Radioactive Materials and Waste" [36].
The following table shows how France has categorized their waste:

Table 5
Classification of radioactive waste in France and disposition plans. CSTFA is the Centre de l'Aube Disposal Facility for VLL Waste and CSFMA is the Centre de l'Aube Disposal Facility for LIL Waste. Source: Nuclear Safety Authority (ASN)96 [36].

<table>
<tr><th>Activity</th><th>Half-life: very short
(less than 100 days)</th><th>Half-life: short
(100 days to 31 years)</th><th>Half-life: long
(more than 31 years)</th></tr>
<tr><td>Very low level (VLL)</td><td rowspan="4">Decay during interim storage</td><td colspan="2">Shallow disposal (CSTFA)</td></tr>
<tr><td>Low level (LL)</td><td rowspan="2">Shallow disposal (CSFMA)</td><td>under study</td></tr>
<tr><td>Intermediate level (IL)</td><td>under study</td></tr>
<tr><td>High level (HL)</td><td colspan="2">(pursuant to 2006 Planning Act)</td></tr>
</table>

80% of high-level waste volumes in France were generated from nuclear power program while another 10% were generated during the defence program (plutonium and tritium production and naval reactors) and research sector. Reprocessing spent fuel's high level waste was stored as La Hague. At La Hague, there are three stores with a combined capacity of 2,174 m^3. AREVA had expanded their capacity to 3,648 m^3 in 2012. The lifetime of this storage is designed to store spent fuel till 2040.

2.3.4.3. Russia

There are two central facilities involved in spent fuel management operate by Rosatom, Russia's Federal Atomic Energy Agency:

- RT-1, the processing plant at Ozersk, has design through put of 400 metric tons per years but has never reprocessed more than 100 metric tons a year [50]. The high-level radioactive waste produced there is verified. An average of 500 tons of verified waste is produced annually.
- RT-2 at Zheleznogorsk was originally intended to be a reprocessing plant but only the spent fuel storage pool was completed before the project stalled during the 1980s. the original design capacity was 13419 VVER-1000 fuel assemblies but has been increased to 7200 tons as result of the installation of higher-density storage racks and additional pool with capacity 0f 1200 tons has been built. As of the end of 2010, more than 6000 tons of spent fuel was stored in the pool. [51].

The pool at the RBMK nuclear power plants also are every close to full. In 2003, therefore, construction of dry spent fuel storage was begun within some of the buildings of the uncompleted RT-2 reprocessing plant. Dry storage capacity of 37785 tons is to be built; with 26510 tons of RMBK-1000 fuel was to be put into operation during 2011 [52].

	Number of units and reactor type[a]	Generating capacity (GWe)	Start of commercial operation	Stored spent fuel (tons)
VVER sites				
Balakovo	4 VVER-1000	3.80	1986 - 1993	400.3
Kalinin	3 VVER-1000	2.85	1985 - 2006	222.1
Kola	4 VVER-440	1.64	1973 - 1984	75.4
Novovoronezh	2 VVER-440	1.77	1972 - 1973	73.9
	1 VVER-1000		1981	138.5
Rostov (Volgodonsk)	2 VVER-1000	1.90	2001	98.2
RBMK sites				
Kursk	4 RBMK-1000	3.70	1977 - 1985	4,612
Smolensk	3 RBMK-1000	2.78	1983 - 1990	2,372
Sosnovy Bor (Leningrad)	4 RBMK-1000	3.70	1974 - 1981	4,485
Other sites				
Beloyarsk	2 AMB[b]			190.9
	1 BN-600	0.56	1981	35.9
Bilibino	4 EGP-6	0.48	1974 - 1976	140.9
Central storage				
Mayak (VVER-440)				379
MCC (VVER-1000)				4,671
Total	31	23.18		17,895.2

Fig. 12. Stored spent fuel in Russian as in 1 January 2008 [53]

2.3.5. Conceptual design of the space disposal system for the highly radioactive component of the nuclear waste

A system for space disposal of the most radioactive component of the spent fuel produced by the nuclear reactor is designed conceptually with intent of ensuring safety. This system is to use a commercially available launch vehicle and be launched from the western Pacific Ocean. It will fly low until it acquires Earth-escape velocity so that the nuclear packages will not fall on land in case of system malfunctioning. The package will be parked in one of lunar liberation point L4 or L5 and kept vigilant by a watch-dog orbital transfer vehicle which will push the package away in case an asteroid approach [54].

2.4. Spent fuel material facility

The spent fuel assemblies are put in a wicker container that is a vital part of the capacity barrel, which is ordinarily fixed utilizing a catapulted top with repetitive seals. Conversely, in the canister-based frameworks, the spent fuel gatherings are set in a thin-walled (normally 12.5 mm or 0.5 in. thick) stainless or carbon steel tube shaped container that is fixed with an internal and an external welded cover. The canister is set in either a barrel shaped cement and steel over pack or a solid vault-sort stockpiling module. The capacity module or over pack ensures the canister against outer characteristic wonders and man-made occasions. The overpack or module is shut with a darted cover or entryway [55]. In the dry storage of spent fuels using concrete casks, stainless-steel canisters act as an important barrier for encapsulating spent fuels and radioactive materials [56].

Fig. 13. Concrete Dry Cask [45]

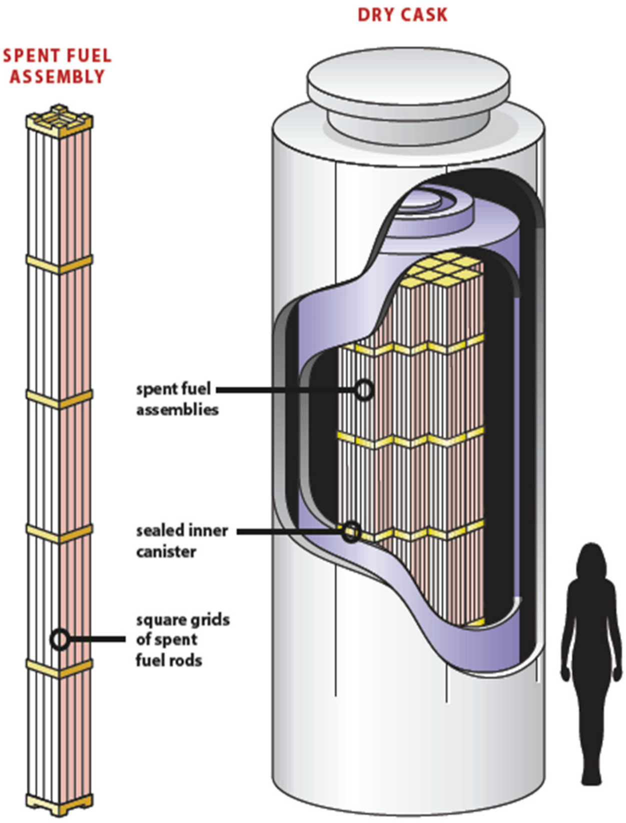

Fig. 14. Illustration of fuel assemblies of dry cask [46]

The storage pool is a reinforced concrete structure usually built above ground or at least at ground elevation. The reinforced concrete structure of the pool, including the covering building, needs to be seismically qualified depending upon national requirements. Most pools are stainless steel lined; some are coated with epoxy resin based paint. In some situations, the pool may be stainless steel lined or epoxy treated only at the water line or at other locations. Regarding unlined and untreated pools properly selected and applied concrete proved to have negligible corrosive ion leaching and permeability to water [57].

3. Conclusion

Spent fuel management has been one of the major problem related to the nuclear power. This is due to the increasing of the nuclear spent fuel over the years. Thus, several ways to manage the spent fuel has been introduced. The method introduced was specially designed for only specified nuclear power plant. Each method introduced has is problem as it may consume a large area and may require to be far away from public due to the high radioactive rate. The improvement in the material aspect for the spent fuel storage was established to reduce the radiation released from spent fuel. More study need to be done in order to achieve the most effective way to manage the spent fuel in order to make is safe to the public and environment by reducing the radiation released.

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