RECOMMENDATIONS FOR THE CONDITIONING OF SPENT METALLIC URANIUM FUEL AND ALUMINIUM CLAD FUEL FOR INTERIM STORAGE AND DISPOSAL

A report prepared by the Technical Committee on Storage and Disposal of Metallic Uranium Fuel and Al-clad Fuels for Nærings- og handelsdepartementet

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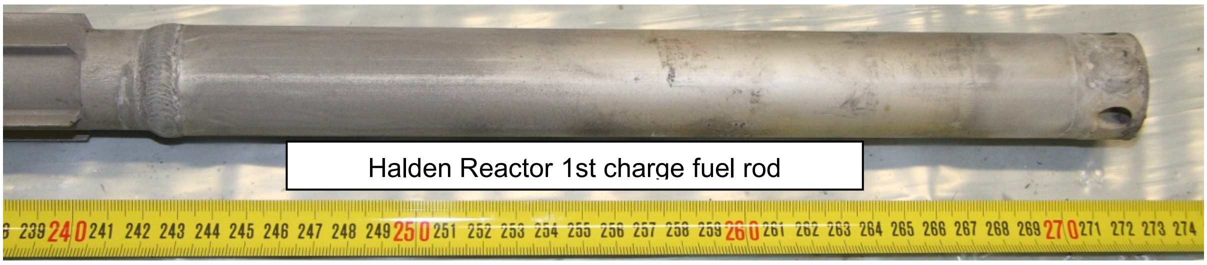

To Nærings- og handelsdepartementet:

This report presents the unanimous recommendations of the Technical Committee on Storage and Disposal of Metallic Uranium Fuel and Al-clad Fuels.

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PREFACE

I would like to express my deep and sincere thanks to all the members of the committee for their contributions to this report. Everyone has made a unique contribution, reflecting their wide range of knowledge and experience. We had long and intense meetings, but with the good humour and willingness of all to help, they were a pleasure to lead.

Thanks are also given to Rolf Jullum (Observer, Statsbygg) for his helpful comments during the past year, and to all the IFE staff who have managed very efficiently the administration of this project - in particular Reidun Engelsrud - and have provided technical information and help when requested, often at very short notice.

I would also like to thank staff from Studsvik Nuclear AB, the IAEA, AREVA NC and Sosny R&D Company for their help and support.

Finally, but not least, special thanks go to our technical secretary, Per Ivar Wethe, IFE, who also acted as a valuable fifth, unofficial member of the committee.

Peter Bennett,

Chairman,

Halden , January 2010.

EXECUTIVE SUMMARY

Background

Spent nuclear fuel (SNF) is irradiated fuel that has been permanently discharged from the reactor. SNF must be stored and/or disposed of in such a way as to isolate it from the environment until the radiotoxicity of the fuel has decayed to levels equal or below that of natural uranium. Storage is taken to mean a process in which SNF can be monitored and retrieved, whereas disposal implies that the SNF is beyond the reach of human intervention.

There are currently three main strategies for the treatment of spent nuclear fuel (the so-called back-end strategy). Firstly, the fuel assemblies can be disposed of in the same physical and chemical form in which they were removed in the reactor, known as “direct disposal”. A second option is “storage and postponed decision”, which is used, for example, before a final decision is made on treatment, further storage and disposal. The third option is chemical treatment or conditioning (commonly known as reprocessing) in which the various components of the SNF are separated and can be stored/disposed of individually - some may be re-used while others can be converted into more stable waste forms. Reprocessing, using the Plutonium - URanium EXtraction (PUREX) process, is the only fuel conditioning process that has been and is currently used on an industrial scale.

There is general endorsement of the concept of geological disposal as the end point for spent fuel. For direct disposal, fuel rods will first be packaged into metal canisters, which will then be placed in the repository, which itself will be back-filled with clay. The use of these various barriers is to ensure that groundwater does not come into contact with the SNF since this could conceivably result in release of radioactive materials to the environment. For most SNF from commercial reactors, the fuel itself acts as the first barrier to retain radioactive material: the fuel is in the form of uranium dioxide (UO_2), a ceramic material that is very stable and almost unreactive towards water. Further, the fuel is clad in Zircaloy, an alloy which is also stable against water and thus forms the second barrier.

SNF in Norway has arisen from the Jeep I and Jeep II reactors at Kjeller, and the Heavy Boiling Water Reactor (HBWR) in Halden. Some of this fuel is similar to commercial fuel and may be suitable for direct disposal in a repository. However, other fuels are chemically reactive and there would be risks associated with direct disposal if they came into contact with groundwater, due to the instability of the material and/or the production of gaseous and potentially explosive reaction products. These materials are fuel manufactured from metallic uranium (U) and fuels clad in aluminium (Al).

Previously, the Bergans committee (1999 – 2001) was appointed to develop a national strategy for disposal of high active, spent nuclear fuel in Norway. Its recommendations were that the decision on a final disposal method should be delayed while awaiting technical developments in other countries, and that immediate work should be started on an intermediate storage facility, in which the SNF should be stored for 50 – 100 years.

Later, in 2004, a Phase 1 committee was established to define Norway's requirements for intermediate storage of HLW; to investigate possible technical solutions; identify critical points with regard to choice of technical solution and location; and to suggest a mandate for a future Phase 2 committee. As technical options, it recommended further investigation of dry storage in a concrete structure or transportable storage containers. It further recommended that the future Phase 2 committee should choose the final technical solution and location of the storage facility. It also specified the need for a technical committee to investigate safe storage of metallic fuel and fuel with Al cladding.

These two new committees were established in January 2009. The Phase-2 committee has the mandate to find the most suitable technical solution and localisation for intermediate storage for spent nuclear fuel and long-lived waste. The Technical Committee on Storage and Disposal of Metallic Uranium Fuel and Al-clad Fuels was given the mandate to recommend treatment options for metallic uranium fuel and aluminium-clad fuel to render them stable for long term storage and disposal. The work of the Technical Committee is recorded in this report.

Inventory of metallic uranium and aluminium-clad fuels

There are approximately 12 tons of metallic uranium / aluminium-clad SNF currently being stored in Halden and Kjeller:

Source (reactor)	Irradiation period	Fuel type	Amount, tonnes
JEEP I	1951 – 1967	Metallic U fuel Al cladding	3
JEEP II	1966 - present	UO_2 fuel Al cladding	2
HBWR 1st charge	1959 – 1960 (reactor still in operation with oxide fuel)	Metallic U fuel Al cladding	7

Chemical and physical properties of metallic uranium and aluminium-clad fuels

The reaction between metallic uranium and oxygen-free water is highly exothermic (i.e. it releases energy to the surroundings), and forms uranium dioxide (UO_2), hydrogen gas (H_2), and uranium hydride (UH_3). If the hydrogen cannot escape, the system pressure will be raised. Metallic uranium and pure uranium hydride are pyrophoric (i.e. they ignite spontaneously in air). The presence of pyrophoric uranium metal and uranium hydride, together with that of hydrogen gas, has resulted in several explosive incidents with metallic uranium fuel in other countries.

The overall reaction between aluminium and water forms aluminium hydroxide ($Al(OH)_3$) and hydrogen gas (H_2). In wet basin storage, pitting has been the main mechanism of aluminium corrosion. Pitting is a localised form of corrosion in which metal is removed preferentially over very small areas on the surface to develop cavities or pits, which results in holes being produced in the metal. It is one of the most destructive and insidious forms of corrosion. The attack is generally limited to extremely small areas, while the remaining surface is relatively unaffected. The pits usually start at small points on the surface and enlarge with time. Pitting is most common on metals covered with an adherent surface film. The pits tend to develop at defects or flaws in the surface film and at sites where the film has been mechanically damaged and does not self-heal. Under certain conditions, the rate of pitting can be rapid and can lead to perforation of the fuel cladding.

International experience and recommendations for treatment of metallic uranium fuels and aluminium-clad fuels

Many countries are participating in the U. S. Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Program and the Russian Research Reactor Fuel Return (RRRFR) program, which allow high enriched spent fuels to be returned to the country of origin. Ownership of the fuel is transferred to either the US or the Russian Federation, along with responsibility for the conditioning and disposition of the fuel. The US plans to reprocess all aluminium fuels at the Department of Energy’s Savannah River Site, whilst fuels returned to Russia will be reprocessed in the Mayak facility. Countries that have shipped SNF in these programs include Argentina, Australia, Austria, Brazil, Canada, Chile, Columbia, Denmark, Finland, Germany, Greece, Indonesia, Italy, Japan, Netherlands, Phillipines, Portugal, Spain, Sweden, Switzerland, Taiwan, Thailand, Uruguay, Venezuela, Bulgaria, the Czech Republic, Kazakhstan, Latvia, Libya, Poland, Romania, Serbia, Uzbekistan and Vietnam.

In other cases, where fuels are not eligible for inclusion in the above programs, countries have purchased commercial reprocessing services in the UK, France and the Russian Federation. Under such contracts, metallic uranium fuel from the Swedish R1 reactor was reprocessed in the UK, while Australian fuels have been treated in France. The US, UK and France either have reprocessed, or plan to reprocess, fuels in domestic facilities.

Following 1976 and 1992 decisions in the US to suspend reprocessing, an extensive program was undertaken to develop alternative methods for conditioning of aluminium fuels. However, the program did not reach maturity for technical and financial reasons, and the current strategy has reverted to reprocessing. Similarly, an investigation was made in Sweden into alternatives to reprocessing the R1 reactor fuel, but a better treatment was not identified.

Commercial (re)processing services are currently offered in France and Russia. The committee has contacted AREVA NC and Sosny Research Company to investigate the possibilities for processing the fuel in, respectively, France and Russia.

Other possible conditioning methods, which are at different stages of development, include electrometallurgical conditioning (pyroprocessing) and calcination. These either require modification to treat the fuels under consideration, or are not technically mature. Further, none are offered commercially.

Storage options for spent fuel and conditioning products include metal/concrete casks, concrete vaults and concrete silo systems. All are in principle suitable for the materials under consideration.

Options for conditioning and assessment criteria

Based on the review of international experience, the following options have been considered for the Norwegian metallic uranium and aluminium fuels:

1. Direct disposal.
2. Long term interim storage with postponed decision.
3. Exchange of fuel.
4. Return to the country of origin.
5. Commercial processing using the PUREX technique.
6. Conditioning in Norway.
    A. Conditioning using developments of PUREX.
    B. Electrometallurgical conditioning (pyroprocessing).
    C. Calcination.

The committee has chosen several criteria against which to assess the above options for conditioning of the fuels:

- The recommendations of the IAEA and OECD NEA, including ethical aspects.
- Technical suitability.
- The maturity and availability of the technology.
- The decisions taken by other countries on similar fuel types.
- Environmental impact.
- Value for money.
- Public acceptance.

The accompanying table summarises the comparison of the options against the selection criteria.

Comparison of options for conditioning of metallic uranium and aluminium-clad spent fuels

Method	**IAEA and OECD NEA recommendations / ethical aspects**	**Technical suitability**	**Technology maturity / availability**	**International experience**	**Environmental impact**	**Value for money**
Direct disposal	Not applicable	Low: possible formation of pyrophoric and gaseous reaction products with water	No experience with type of fuels under consideration. Concept accepted in, among others, Sweden and Finland for ***oxide*** fuel.	None	Poor: possible formation of pyrophoric and gaseous reaction products with water. Possible release of radioactivity to the biosphere.	Good
Storage and postponed decision	Ethical objections	Low: degradation of fuel may occur	Mature: current situation	Extensive	Poor: degradation of fuel will occur	Poor: is not a final solution
Exchange of fuel	Ethical objections	Unknown: no potential exchange identified	Unknown: no potential exchange identified	Cases exist.	Unknown: no potential exchange identified	Unknown: no potential exchange identified
Return to the country of origin	Supported: eg US and Russian Return Programs	Mature and well-proven	Mature and well-proven	Mature and well-proven	Waste products stable in repository (fuel will be conditioned by PUREX).	Good
Commercial processing using PUREX	Supported: eg Serbian fuel, US and Russian Return Programs	Mature and well-proven	Mature and well-proven	Mature and well-proven	Waste products stable in repository (fuel will be conditioned).	Good: commercial services offered.

Method	**IAEA and OECD NEA recommendations / ethical aspects**	**Technical suitability**	**Technology maturity / availability**	**International experience**	**Environmental impact**	**Value for money**
Domestic conditioning using development of PUREX	Not applicable	PUREX is mature and well-proven. Developments underway internationally	PUREX is mature and well-proven. Developments underway internationally	PUREX is mature and well-proven. No experience with developments	Waste products stable in repository (fuel will be conditioned). Additional radioactive waste from domestic facility	Very poor: domestic facility required
Domestic conditioning using electrometallurgical conditioning	Not applicable	Medium: requires extra step for Al removal	Clad removal step unproven	Metallic U fuel with stainless steel cladding (EBR-II fuel)	Waste products probably stable in repository	Very poor: domestic facility required
Domestic conditioning using calcination	Not applicable	Low: requires decladding and extensive cutting of fuel rods, plus further conditioning to produce a stable waste form.	Not proven	Tested on unirradiated material	Waste products would be stable in repository if suitable conditioning method identified.	Very poor: domestic facility required

The Technical Committee's recommendations

After assessing the different options for spent fuel conditioning against the selection criteria, the following recommendations are made:

1. Metallic uranium is pyrophoric, and reacts with water to form hydrogen gas and pyrophoric uranium hydride. Aluminium cladding reacts with water to form hydrogen gas, and can undergo severe localised corrosion, which can penetrate the cladding and expose the metallic uranium fuel. Since metallic uranium fuel and/or fuel clad in aluminium are unstable, they should be stabilised before interim storage and disposal. The method chosen to stabilise the fuels should render them eligible for both interim storage and consequent disposal: i.e. such that further treatment of the fuel after interim storage is not required.

2. The storage and postponed decision or "wait and see" option does not constitute an endpoint for the fuel. The committee considers that this option represents a major uncertainty/deficiency since it presumes stability of future societies and their continuing ability to carry out the required safety and institutional measures, and since Norway is not working actively to investigate or develop alternative solutions, such as solutions based on technology development or multi-national facilities. In the light of these objections, and in-line with the Swedish decision on management of the R1 reactor fuel, the committee recommends against this option for the Norwegian spent fuel.

3. Similar objections apply to consideration of exchanging fuel with another country, as the responsibility for managing the fuel is transferred. The committee thus recommends against this option for the Norwegian spent fuel.

4. The small volume of spent nuclear fuel in Norway makes it difficult to justify the costs of constructing a domestic conditioning facility as long as overseas commercial services are available that satisfy financial, political, and environmental requirements.

5. The committee recommends that the Norwegian government immediately makes a high-level contact with the United States government to discuss the possibility of shipping the fuel to the United States under the current take back programs since these will cease in 2016.

6. The committee recommends that the Norwegian government sends to Rosatom State Corporation a letter of request to consider the possibility of importing the spent fuel into the Russian Federation for processing. Following the 2009 report of the IAEA Contact Expert Group on management of spent nuclear fuel and radioactive wastes, and in view of the fact that the Norwegian government is donating funds which contribute to processing of Russian fuel in Mayak, the committee considers that this option satisfies environmental and political requirements. The committee recommends that, if a decision is taken to condition the fuel using Russian commercial services, it should be taken as early as possible in order to take advantage of the current availability of these services. This availability is linked to the activities of the present Russian take back program.

7. The committee recommends that the Norwegian government take the necessary steps to acquire a report from AREVA NC in France summarising the feasibility of processing the spent fuel in the Cap La Hague facilities. The report should include technical feasibility, cost and options for management of the processing products.

8. The decision on the method of interim storage of the conditioning products following treatment of the metallic fuel should be taken after the decision on the stabilisation method. The choice of storage method should be based on the physical, chemical and radiological properties of the waste form(s) together with an economical analysis of the three options. The committee recommends dry storage in casks, silos or vaults.

Concluding remarks and next steps

Originally, it was envisaged that there would be a range of available treatment options, which could be compared with each other and for which the volumes and types of material to be stored could be calculated. However, in practice, once recommendations were made against direct disposal and interim storage with postponed decision, one treatment option – overseas commercial processing – is clearly shown to be superior when compared against the chosen selection criteria.

Thus, the next step should be a high level political decision on whether to pursue the overseas commercial processing option. If this option is to be progressed, a decision must also be taken on whether to take back the conditioning products where there is an option to dispose of them in the processing country. For processing at Mayak, there are options to store and dispose of all waste products in Russia, whilst French law prescribes return of high level waste. Some of the countries participating in the Russian Research Reactor Fuel Return Program and all of the countries participating in the U. S. Foreign Research Reactor Spent Nuclear Fuel Acceptance Program will store and dispose of all waste products in Russia and the U. S., respectively.

If, instead, a domestic conditioning treatment is chosen, a decision on the method will be required (for example, wet processing using PUREX or calcination). In all cases, approximately around 20 tons (10 m^3) of uranium product and 1.5 kg of plutonium would be produced, but the high level, secondary and decommissioning wastes would be dependent on the method.

Only when the waste forms and volumes are known can recommendations be made for storage methods; and, as discussed above, it is necessary first to take a fundamental decision regarding the conditioning method.

SAMMENDRAG

Bakgrunn

Brukt kjernebrensel er bestrålt brensel som er tatt ut av reaktorkjernen og ikke skal brukes på nytt. Brukt brensel må lagres og/eller deponeres på en slik måte at det isoleres fra naturmiljøet inntil radiotoksisiten til det brukte brenselet har avtatt til et nivå som tilsvarer eller er lavere enn radiotoksisiteten til naturlig uran. Lagring betyr her en prosess der brukt kjernebrensel kan bli overvåket og tatt tilbake; mens deponering innebærer at det brukte brenselet er utenfor rekkevidde fra menneskelige inngrep.

Det eksisterer i dag tre hovedstrategier for behandling av brukt kjernebrensel (sluttfasestrategier). For det første kan brenselet bli deponert i den samme fysiske og kjemiske form som det hadde i reaktorkjernen, såkalt ”direkte deponering”. En annen opsjon er ”lagring og utsatt beslutning”, som er brukt, for eksempel, før det tas en beslutning om behandling, videre lagring og deponering. Den tredje opsjonen er kjemisk behandling (vanligvis kjent som reprosessering) slik at de forskjellige komponentene av det brukte brenselet blir separert og kan lagres/behandles individuelt – noe kan brukes på nytt, mens andre komponenter kan omdannes til mer stabile avfallsformer. Reprosessering, ved bruk av Plutonium – URanium EXtraction (PUREX) prosessen, er den eneste prosessen for behandling av brukt kjernebrensel som har vært brukt og brukes idag i industriell skala.

Geologisk deponering av brukt brensel er i dag en akseptert teknisk løsning som har bred internasjonal faglig støtte. Når det gjelder direkte deponering av brukt brensel, vil det brukte brenslet først bli kapslet inn i tette metallbeholdere som deretter blir plassert i deponiet. Selve deponiet blir i sin tur fylt igjen med leire. Disse forskjellige barrierene skal hindre og forsinke en eventuell kontakt mellom det brukte brenselet og grunnvann, og dermed spredning av radioaktive materialer til miljøet. Når det gjelder brukt brensel fra kommersielle kjernekraftverk, er det i de fleste tilfeller brenselet selv som utgjør den første barrieren mot spredning av radioaktivitet: brenselet er urandioksid (UO_2) , et keramisk materiale som er meget stabilt og nesten uløselig i vann. Videre er selve brenselet kapslet inn i Zircaloy, en legering som er meget stabil i vann og således utgjør den andre barrieren.

Brukt brensel i Norge skriver seg fra driften av JEEP I og JEEP II reaktorene på Kjeller, og HBWR (Halden Boiling Water Reactor) i Halden. Noe av dette brenselet er av samme type som kommersielt kjernebrensel og kan deponeres direkte i et deponi. Det øvrige brenselet er ustabilt, dvs. kjemisk reaktivt, og representerer en risiko hvis det skulle komme i kontakt med grunnvann; da det kan dannes gassformige og potensielt eksplosive reaksjonsprodukter. Dette gjelder for brenselsmaterialer som inneholder metallisk uran (U) og brenselsmaterialer med en kapsling av aluminium (Al).

På slutten av 1990-årene ble Bergans-komitéen(1999-2001) etablert for å utvikle en nasjonal strategi for sluttbehandling og deponering av høy aktivt, brukt kjernebrensel i Norge. Bergan-komitéen anbefalte at beslutningen vedrørende en endelig deponeringsmetode burde utsettes i påvente av tekniske utviklingsarbeider i andre land, og at det umiddelbart burde igangsettes arbeid med et mellomlager, der det brukte brenselet kunne lagres i 50-100 år.

Noe senere, i 2004, ble Fase 1-komitéen etablert for å få definert de norske kravene til et mellomlager for høyaktivt radioaktivt avfall, undersøke mulige tekniske løsninger, identifisere kritiske momenter med hensyn til valg av tekniske løsninger og lokalisering av et slikt lager, og foreslå et mandat for en fremtidig Fase 2 komité. Som tekniske løsninger anbefalte Fase 1-komitéen videre undersøkelser av tørr lagring i en betongstruktur eller transportable lagringsbeholdere. Fase 1-komitéen anbefalte også at den fremtidige Fase 2-komitéen skulle velge teknisk løsning og lokalisering av mellomlageret. Fase 1-komitéen klargjorde også behovet for en Teknisk komité for å utrede sikker lagring av metallisk uranbrensel og brensel med aluminiumskapsling.

Disse to nye komitéene ble etablert i januar 2009. Fase 2-komitéen skal i følge sitt mandat anbefale den mest hensiktsmessige tekniske løsning og lokalisering for et mellomlager for brukt brensel og lang-livet avfall. Det ”tekniske komité for lagring og deponering av metallisk uran og brensel med aluminiumskapsling” skal i følge sitt mandat anbefale behandlingsmetoder for stabilisering av metallisk uranbrensel og brensel med aluminiumskapsling slik at dette kan lagres og deponeres. Denne rapporten beskriver resultatet av det arbeidet som er utført av Teknisk komité.

Mengde metallisk uranbrensel og brensel med aluminiumskapsling

Det er per i dag lagret ca. 12 tonn metallisk uran/aluminiumskapslet brukt kjernebrensel i Halden og på Kjeller:

Kilde (reactor)	**Bestrålingsperiode**	**Brenselstype**	**Mengde, tonn**
JEEP I	1951 – 1967	Metallisk uranbrensel med aluminiumskapsling	3
JEEP II	1966 – til I dag	UO_2 brensel med aluminiumskapsling	2
HBWR 1. ladning	1959 – 1960 (reactor fortsatt I drift med oksidbrensel)	Metallisk uranbrensel med aluminiumskapsling	7

Kjemiske og fysiske egenskaper ved metallisk uran og brensel med aluminiumskapsling

Reaksjonen mellom metallisk uran og oksygenfattig vann er *kraftig* eksoterm, dvs. den frigjør energi til omgivelsene. Ved reaksjonen dannes urandioksid (UO_2), hydrogengass (H_2) og uranhydrid (UH_3). Hvis hydrogengassen ikke kan slippe ut av systemet vil systemtrykket øke. Metallisk uran og rent uranhydrid er pyrofort, dvs. antennes spontant i luft. Tilstedeværelsen av pyrofort uranmetall og uranhydrid, sammen med hydrogengass, har ført til en rekke eksplosjonsuhell med metallisk uranbrensel i andre land.

Totalreaksjonen mellom aluminium og vann fører til dannelse av aluminiumhydroksid ($Al(OH)_3$) og hydrogengass (H_2). Korrosjon av aluminiumskapslet brukt kjernebrensel er imidlertid avhengig av flere innbyrdes avhengige faktorer. Ved våtlagring av brukt brensel har groptæring vært hovedmekanismen ved aluminiumskorrosjon. Groptæring er et lokal, punktformet korrosjonsangrep. Metallet angripes og løses opp i meget små områder på overflaten slik at det dannes hulrom eller groper som resulterer i gjennomgående hull og fullstendig ødeleggelse av metallet. Dette er en av de mest ødeleggende og *farlige* former for korrosjon da korrosjonsangrepet i alminnelighet er begrenset til ekstremt små områder, mens resten av overflaten er relativt upåvirket. Gropene starter vanligvis ved små punkter på overflaten og vokser med tiden. Groptæringen forekommende gjerne på metallflater med et tynt overflatebelegg, som for eksempel en tynn oxidfilm. Gropene starter vanligvis ved defekter eller feil i overflatefilmen og på steder der filmen er blitt mekanisk skadet og den ikke reparerer seg selv. Under visse betingelser kan groptæringen og gjennombrudd av brenselskapslingen utvikles svært hurtig .

Internasjonale erfaringer og anbefalinger for behandling av metallisk uranbrensel og brensel med aluminiumskapsling

Mange land deltar i det amerikanske "U.S. Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Program" og det russiske "Russian Research Reactor Fuel Return (RRRFR) Program", som tillater at høy-anriket brukt brensel blir returnert til opprinnelseslandet. Eierskapet til brenselet overføres enten til den amerikanske eller den russiske regjeringen, sammen med ansvaret for behandling og deponering av brenselet. Amerikanerne planlegger å reprosessere alt aluminiumskapslet brensel ved Energidepartementets Savannah River anlegg, mens brensel som returneres til Russland vil bli reprosessert ved Mayak-anlegget. Følgende land har benyttet seg av disse returmulighetene: Argentina, Australia, Brasil, Canada, Chile, Colombia, Danmark, Filippinene, Finland, Hellas, Indonesia, Italia, Japan, Nederland, Portugal, Spania, Sverige, Sveits, Taiwan, Thailand, Tyskland, Uruguay og Venezuela, Bulgaria, Kazakhstan, Latvia, Libya, Polen, Romania, Serbia, Tsjekkia, Uzbekistan og Vietnam.

I andre tilfeller, der brenslet ikke kvalifiseres for en behandling under de ovennevnte amerikanske eller russiske rammeprogrammene, har eierlandene kjøpt kommersielle reprosesseringstjenester i Frankrike, i Storbritannia og i Russland. Under en slik kontrakt ble metallisk uranbrensel fra den svenske R1 reaktoren reprosessert i Storbritannia, mens australsk brensel har vært behandlet i Frankrike. USA, Storbritannia og Frankrike har enten reprosessert eller planlegger å reprosessere brensel i egne anlegg.

Etter at USA i 1976 og 1992 besluttet å innstille reprosessering, ble det satt i gang et omfattende program for å utvikle alternative metoder for behandling av aluminiumskapslet brensel. Programmet har, av tekniske og finansielle årsaker, ikke nådd målsettingen og dagens strategi i USA har vendt tilbake til reprosessering. I Sverige ble det gjort en tilsvarende utredning omkring alternativer til reprosessering av R1 kjernebrenselet, men det ble ikke funnet noe bedre behandlingsalternativ en reprosessering.

Frankrike og Russland tilbyr i dag kommersielle (re)prosesseringstjenester. Kommitteen tok kontakt med AREVA NC og Sosny Research Company for å undersøke mulighetene for prosessering av brenselet i henholdsvis Frankrike og Russland.

Andre mulige behandlingsmetoder, som befinner seg på ulike tekniske utviklingstrinn , omfatter elektrometallurgisk behandling (pyroprosessering) og kalsinering. Disse metodene må enten modifiseres for å kunne behandle det aktuelle brenselet, eller er ikke teknisk utviklet. Videre, er ingen av disse metodene kommersielt tilgjengelig.

Aktuelle alternativer for lagring av brukt brensel og stabiliserte sluttprodukter fra brenselsbehandlingen omfatter metall/betongbeholdere, betonghvelv og betongsilo systemer. Alle alternativene er i prinsippet egnet for de materialene som her er aktuelle.

Behandlingsalternativer og vurderingskriterier

Etter en gjennomgang av de internasjonale erfaringene er følgende alternativer vurdert for det norske metalliske uranbrenselet og brensel med aluminiumskapsling:

1. Direkte deponering.
2. Langtids-mellomlagring med utsatt beslutning.
3. Bytte av brensel.
4. Retur til opprinnelseslandet.
5. Kommersiell brenselsbehandling med PUREX teknikk.
6. Behandling i Norge.
    A. Behandling med videreutviklet PUREX.
    B. Elektrometallurgisk behandling (pyroprosessering).
    C. Kalsinering.

Komitéen har valgt følgende kriterier i vurdering av ovennevnte alternativer for brenselsbehandlingen:

- Anbefalingene fra IAEA og OECD/NEA, innbefattet etiske aspekter.
- Teknisk velegnethet.
- Teknologisk modenhet og tilgjengelighet.
- Andre lands beslutninger knyttet til samme typer av brensel.
- Miljøpåvirkninger.
- Kostnadseffektivitet.
- Offentlig aksept/godkjennelse.

Følgende tabell gir en oppsummering av de ovennevnte alternativene vurdert i lys av utvalgskriteriene.

Sammenligning av alternativer for behandling av metallisk uran og brensel med aluminiumskapsling

Metode	**IAEA og OECD/ NEA anbefalinger / etiske aspekter**	**Teknisk hensiktsmessighet**	**Teknologisk modenhet og tilgjengelighet**	**Internasjonal erfaring**	**Miljøpåvirkning**	**Kostnadseffektivitet**
Direkte deponering	Ikke anvendelig/egnet	Lav:mulig dannelse av pyrofore (selvantennelige)og gassformige reaksjonsprodukter med vann	Ingen erfaring med slike brenselstyper som her er aktuelle. For oksid brensel er konseptet akseptert i bl.a Sverige,og Finland	Ingen	Dårlig: mulig dannelse av pyrofore (selvantennelige) og gassformige reaksjonsprodukter med vann. Mulig frigjøring av radioaktivitet til biosfæren.	God
Lagring og utsatt beslutning	Etiske innvendinger	Lav: nedbrytning av brenselet kan forekomme	Moden: dagens situasjon	Omfattende	Dårlig:nedbrytning av brenselet vil forekomme	Dårlig: er ikke et behandlingsteknisk slutt-trinn
Utveksling av brensel	Etiske innvendinger	Ukjent: ingen potensielle muligheter er identifisert	Ukjent: ingen potensielle muligheter er identifisert	Enkelte tilfeller har forekommet	Ukjent: ingen potensielle uligheter er identifisert	Ukjent: ingen potensielle muligheter er identifisert
Retur til opprinnelselandet	Støttes:f.eks.amerikansk og russisk returprogram	Modent og velprøvd	Modent og velprøvd	Modent og velprøvd	Avfallsprodukter stabile i deponi (brensel vil bli behandlet med PUREX teknikk	God
Kommersiell prosessering med PUREX teknikk	Støttes:f.eks.serbisk brensel, amerikansk og russisk returprogram	Modent og velprøvd	Modent og velprøvd	Modent og velprøvd	Avfallsprodukter stabile i deponi (brensel vil bli behandlet)	God: kommersielle tjenester er tilgjengelig

Metode	**IAEA og OECD/ NEA anbefalinger / etiske aspekter**	**Teknisk hensiktsmessighet**	**Teknologisk modenhet og tilgjengelighet**	**Internasjonal erfaring**	**Miljøpåvirkning**	**Kostnadseffektivitet**
Innenlandsk behandling med videreutviklet PUREX teknikk	Ikke anvendbart	Purex er moden og velprøvd. Videreutvikling pågår internasjonalt.	PUREX er moden og velprøvd. Videreutvikling pågår internasjonalt.	PUREX er moden og velprøvd. Ingen erfaring med videreutvikling.	Avfallsprodukter stabile I deponi(brensel vil bli behandlet). Ytterlig radioaktivt avfall fra innenlandsk anlegg.	Meget dårlig: nødvendig med innenlandsk anlegg
Innenlandsk behandling med elektrometallurgisk behandling	Ikke anvendbar	Middels: krever ekstra prosesstrinn for fjerning av aluminium	Fjerning av kapslingsmateriale. Uprøvd prosesstrinn	Metallisk U brensel med kapsling av rustfritt stål(EBR-II brensel)	Avfallsprodukter sannsynlig stabile I deponi	Meget dårlig: nødvendig med innenlandsk anlegg
Innenlandsk behandling med kalsinering	Ikke anvendbar	Lav: krever fjerning av kapsling og omfattende oppkapping av brenselet, samt ytterligere behandling for å fremstille et stabilt avfallsprodukt	Ikke utprøvd	Testet på ikke-bestrålt materiale	Avfallsprodukter ville bli stabile i deponi hvis hensiktsmessig behandlingsmetode identifiseres	Meget dårlig: nødvendig med innenlandsk anlegg

Teknisk Komités anbefalinger

Etter en totalvurdering av de ovennevnte alternativene i lys av utvalgskriteriene, gir Teknisk Komité følgende anbefalinger:

1. Metallisk uran er pyrofort (selvantennelig i luft) og reagerer med vann slik at det dannes hydrogengass og pyrofort uranhydrid. Aluminiumskapsling reagerer med vann og danner hydrogengass og kan dessuten utsettes for en omfattende, alvorlig lokal korrosjon som kan ødelegge kapslingen og frilegge det metalliske uranbrenselet. Siden metallisk uranbrensel og/eller brensel med aluminiumskapsling er ustabilt, må dette brenselet stabiliseres før mellomlagring og deponering. Den metoden som velges for stabilisering må klargjøre brenselet både for mellomlagring og påfølgende deponering, dette innebærer at videre behandling av brenselet etter mellomlagring ikke er påkrevet, dvs . at brenselet er deponeringsklart.

2. Lagring og utsatt beslutning eller "vente og se" alternativet representerer ikke noe sluttresultat for brenselet. Teknisk komité anser at dette alternativet innebærer en stor svakhet siden det forutsetter stabilitet i fremtidige samfunn, og disse samfunns vedvarende evne og mulighet til å iverksette og forvalte de nødvendige sikkerhets - og institusjonelle tiltak, og siden Norge ikke arbeider aktivt med å utforske eller utvikle alternative løsninger, for eksempel alternativer basert på teknologisk utvikling eller flernasjonale anlegg. I lys av disse innvendingene, og i overensstemmelse med den svenske beslutningen vedrørende R1 kjernebrenselet, vil Teknisk komité fraråde vente og se alternativet når det gjelder behandling av det norske kjernebrenselet.

3. Lignende innvendinger gjelder også for utveksling/bytte av brukt brensel med et annet land, siden ansvaret for brenselet fraskrives. Teknisk komité vil derfor fraråde dette alternativet når det gjelder behandling av det norske kjernebrenselet.

4. Det beskjedne volumet av brukt reaktorbrensel i Norge gjør det vanskelig å begrunne bygging av et norsk anlegg for behandling av brukt brensel så lenge det finnes tilgjengelige utenlandske kommersielle tjenester som tilfredsstiller økonomiske, politiske og miljømessige krav.

5. Teknisk komité anbefaler at Norge umiddelbart, og på regjeringsnivå, tar kontakt med USA for å drøfte transport av brenselet til USA innenfor de eksisterende returprogrammene, siden disse vil avsluttes i 2016.

6. Teknisk komité anbefaler at den norske regjering retter et skriftlig forespørsel til Rosatom State Corporation i Russland for å vurdere muligheten for å importere brukt brensel til Russland for prosessering. Med henvisning til 2009 rapporten fra IAEA "Contact Expert Group on management of spent nuclear fuel and radioactive wastes" og i lys av det faktum at den norske regjering gir økonomiske midler som bidrar til prosessering av russisk brensel i Mayak, anser komitéen at dette alternativet tilfredstiller miljømessige og politiske krav. Teknisk komité anbefaler, at hvis det besluttes å bruke russiske kommersielle tjenester for behandling av brenselet, bør en slik beslutning treffes så tidlig som mulig for å kunne utnytte den aktuelle tilgjengeligheten av disse tjenestene.. Denne tilgjengeligheten er knyttet til virksomhetene i det nåværende russiske returprogrammet.

7. Teknisk komité anbefaler at den norske regjering treffer de nødvendige tiltak for å få AREVA NC i Frankrike til å utarbeide en rapport som beskriver mulighetene for behandling av det brukte brenselet i Cap de La Hague anlegget. Denne rapporten bør omfatte teknisk gjennomførbarhet, kostnader og alternativer for behandling av de ulike prosessproduktene.

8. Beslutningen som angår mellomlagring av de prosessproduktene som oppstår ved behandling av det metalliske brenselet, bør treffes i lys av den valgte stabiliseringsmetoden. Den valgte lagringsmetoden bør baseres på de fysiske, kjemiske og radiologiske egenskapene ved avfallsformen(e), samt en økonomisk analyse av de tre alternativene. Teknisk komité anbefaler tørr lagring i beholdere, siloer eller i hvelv.

Avsluttende kommentarer og videre fremdrift

I utgangspunktet har man sett for seg at det kunne finnes en rekke tilgjengelige behandlingsalternativer som lot seg sammenligne innbyrdes, og der volumer/mengder og typer av de materialene som skulle lagres, kunne tallfestes. I praksis har det imidlertid vist seg, at siden Teknisk komité ikke kan anbefale direkte deponering av det brukte brenselet eller mellomlagring med utsatt beslutning, fremstår følgende behandlingsalternativ – kommersiell prosessering i et utenlandsk anlegg – som det overlegent beste alternativet i lys av de valgte kriteriene.

Neste trinn bør således være en politisk beslutning på høyt nivå, om kommersiell prosessering i et utenlandsk anlegg, skal følges opp. Hvis dette alternativet blir valgt må det også treffes en beslutning om enten å ta tilbake prosessproduktene eller la disse bli i det aktuelle landet, såfremt dette er et alternativ i vedkommende land. For prosessering i Mayak er det muligheter for lagring og deponering av alle avfallsproduktene i Russland, men fransk lov påbyr retur av høyaktivt avfall. Noen av de land som deltar i det russiske returprogrammet og alle de land som deltar i det amerikanske returprogrammet for brukt brensel fra forskningsreaktorer, vil lagre og deponere avfallsproduktene i henholdsvis Russland og USA.

Hvis det derimot velges en nasjonal behandlingsstrategi, må det treffes en beslutning om metode (f.eks. våtkjemisk prosessering basert på PUREX eller kalsinering). Uansett metode vil det bli produsert ca 20 tonn (10m^3) uranprodukter og 1,5 kg plutonium. Mengden av høyaktivt avfall, sekundært avfall og avfall knyttet til nedlegging en av aktuelle anlegg vil imidlertid bli bestemt av den behandlingsmetoden som blir valgt.

Bare når de aktuelle avfallstyper og avfallsvolumer er kjente størrelser, kan det besluttes om lagringsmetoder. Som drøftet ovenfor må det først treffes en grunnleggende beslutning om hvilken behandlingsmetode som skal velges.

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1 DEFINITIONS AND ABBREVIATIONS

1.1 Definitions

For the purpose of this report, the definitions below are adopted, which have been taken from the IAEA's Radioactive Waste Management Glossary, 2003 edition [IAEA, 2003a].

Conditioning. Those operations that produce a waste package suitable for handling, transport, storage and/or disposal. Conditioning may include the conversion of the waste to a solid waste form.

Disposal. The emplacement of spent fuel or radioactive waste in an appropriate facility without the intention of retrieval.

Dry storage. The storage of fuel assemblies and fuel elements and related components in a gas environment, such as air or an inert gas.

Fuel assembly (FA). Fuel elements and associated components that are installed as a single unit in the reactor core and are not disassembled during installation and removal from the reactor core.

Fuel element. A component of the fuel assembly that consists primarily of the nuclear fuel and its cladding materials; for example, individual rods, tubes or plates of an assembly.

HEU (high enriched uranium). Uranium enriched to a level of at least 20 per cent of the isotope ^{235}U.

High level waste (HLW). The radioactive liquid containing most of the fission products and actinides present in spent fuel – which forms the residue from the first solvent extraction cycle in reprocessing – and some of the associated waste streams; this material following solidification; spent fuel (if it is declared a waste); or any waste stream with similar radiological characteristics. Typical characteristics of HLW are thermal powers of about 2 kW/m^3 and long lived radionuclide concentrations exceeding the limits for short lived waste.

Interim storage. The storage of spent fuel and related components such that isolation, monitoring, environmental protection and human control are provided until it is retrieved for further processing or direct disposal.

LEU (low enriched uranium). Uranium enriched to a level below 20 per cent of the isotope ^{235}U.

Pretreatment. Any or all of the operations prior to waste treatment, such as collection, segregation, chemical adjustment and decontamination.

Processing. Any operation that changes the characteristics of waste, including pretreatment, treatment and conditioning.

PUREX process. The reprocessing of spent fuel by dissolving it in nitric acid and separating the uranium and plutonium from fission and activation products by extraction with an organic solvent.

Radioactive waste. Radioactive material in gaseous, liquid or solid form for which no further use is foreseen and which is controlled as radioactive waste by the regulatory authority under the legislative and regulatory framework.

Spent fuel or spent nuclear fuel (SNF). Nuclear fuel that has been irradiated in and permanently removed from a reactor core regardless of burn-up.

Spent fuel conditioning. A special processing operation to prepare spent fuel for disposal.

Spent fuel management. All the activities, administrative and operational, that involve the handling, storage, transport and processing of spent fuel, as well as the storage, transport and disposal of its derivatives, performed to guarantee the safety and security of the fuel. It may also involve discharges.

Spent fuel processing. Either spent fuel conditioning or spent fuel reprocessing.

Spent fuel reprocessing. A process or operation, the purpose of which is to extract radioactive isotopes from spent fuel for further use.

Spent fuel storage facility. An installation used for the interim storage of fuel assemblies, fuel elements and related components after their removal from the reactor and before their processing or disposal as radioactive waste.

Storage. The holding of spent fuel or radioactive waste in a facility that provides for its containment, with the intention of retrieval.

Treatment. Operations intended to benefit safety and/or economy by changing the characteristics of the waste. Three basic treatment objectives are volume reduction, removal of radionuclides and change of composition. Treatment may result in an appropriate waste form.

Waste form. Waste in its physical and chemical form after treatment and/or conditioning (resulting in a solid product) prior to packaging. The waste form is a component of the waste package.

Waste package. The product of conditioning that includes the waste form and any container(s) and internal barriers (e.g. absorbing materials and liners), prepared in accordance with the requirements for handling, transport, storage and/or disposal.

Wet storage. Storage of spent fuel in water.

1.2 Abbreviations

AECL. Atomic Energy of Canada Limited.

AFR. Away From Reactor (refers to the location of storage facilities).

ANSTO. Australian Nuclear Science Technology Organisation.

AR. At Reactor (refers to the location of storage facilities).

AREVA NC. French-based nuclear fuel cycle company.

BNFL. British Nuclear Fuels Limited.

BRR. Budapest Research Reactor.

CANDU. Canada Deuterium Uranium reactor.

CASTOR. Spent fuel storage casks from GNB, Germany.

CLAB. Centralt mellanlager för använt kärnbränsle (Swedish for 'Central holding storage for spent nuclear fuel').

CoRWM. Committee on Radioactive Waste Management (UK).

CRL. Chalk River Laboratory, Canada.

DOE. (US) Department of Energy.

DSF. Dry Storage Facility.

DWPF. Defence Waste Processing Facility (USA).

EBR-II. Experimental Breeder Reactor II (USA).

EDF. Electricité de France.

FRRSNF. (The U. S.) Foreign Research Reactor Spent Nuclear Fuel Acceptance Program.

GNEP. Global Nuclear Energy Partnership.

GRTI. Global Threat Reduction Initiative (USA).

HABOG. Multi-function dry storage vault (Netherlands).

HBWR. Halden Boiling Water Reactor.

HIFAR. High Flux Australian Reactor.

HEU. High Enriched Uranium.

INEEL. Idaho National Engineering and Environmental Laboratory.

ISF. Idaho Spent Fuel Project.

JEEP I / JEEP II. Research reactors at IFE, Kjeller, Norway.

IAEA. International Atomic Energy Agency.

LEU. Low Enriched Uranium.

MAGNOX. UK gas-cooled reactor.

MVDS. Modular Vault Dry Storage.

MTR. Material Test Reactor.

NDA. Nuclear Decommissioning Authority (UK).

NORA. Research reactor at IFE, Kjeller, Norway

NPP. Nuclear Power Plant.

NTI. Nuclear Threat Initiative (US).

NUHOMS. Spent fuel storage system.

OECD NEA. Organisation for Economic Cooperation and Development Nuclear Energy Agency.

OPAL. Open Pool Australian Light Water reactor.

PUREX. Plutonium - URanium EXtraction — the standard aqueous method for the recovery of uranium and plutonium from used nuclear fuel.

RERTR. Reducement Enriched Research and Test Reactor Program.

Rosatom. Russian state nuclear energy company.

RRRFR. Russian Research Reactor Fuel Return Program.

RRSNF. Research Reactor Spent Nuclear Fuel.

SFA. Spent fuel assembly.

SKB. Swedish Nuclear Fuel and Waste Management Company.

SNF. Spent Nuclear Fuel.

SKI. Statens Kärnkraftinspektion.

Sosny. Russian research and development company specialising in spent nuclear fuel.

SSI. Swedish Radiation Safety Authority.

SRS. Savannah River Site (USA).

Studsvik Nuclear AB. Swedish research and development company offering technical services to the international nuclear power industry.

SVAFO (AB). Swedish non-commercial company responsible for managing historic radioactive waste.

THORP. Thermal Oxide Reprocessing Plant (UK).

1.3 Commentary

According to the above definitions and the mandate of the Technical Committee (see Chapter 2.3.3), the task of the committee was to recommend methods for the *conditioning* of spent fuel. The differences between the definitions of spent fuel conditioning and spent fuel reprocessing differ only in that the latter implies that further use is planned for all or part of the separated components of the spent fuel; the actual process employed can be identical in both cases. In addition, the above definitions are not used universally, with the result that confusion can arise as to the exact meaning of the terminology used in some reports. This situation is exacerbated by the common identification of the general term reprocessing with the specific PUREX process, which has arisen because PUREX is the only reprocessing technique used on an industrial scale. In the sections of this report that present the results of literature surveys, the terms used in the original reports are kept even though in many cases it may be more appropriate to change the text according to the above definitions.

2 BACKGROUND AND INTRODUCTION

2.1 Background

Spent nuclear fuel (SNF) is defined as irradiated fuel that has been permanently discharged from the reactor and is not planned to be irradiated further. SNF is highly radioactive due to the fission products and trans-uranic elements formed during irradiation. This radioactivity decreases with time as the radioactive elements decay to stable material. Thus, SNF must be stored and/or disposed of in such a way as to isolate it from the environment until the radiotoxicity of the fuel has decayed to levels equal or below that of natural uranium. Storage is taken to mean a process in which SNF can be monitored and retrieved; whereas disposal implies that the SNF is beyond the reach of human intervention. Interim storage is considered to be for periods of up to approximately 100 years.

There are currently three main strategies for the treatment of spent nuclear fuel (the so-called back-end strategy). Firstly, the fuel assemblies can be disposed of in the same physical and chemical form in which they were removed in the reactor, known as “direct disposal” or “once through fuel cycle”. A second option is “storage and postponed decision” or the “wait and see option”, which is used, for example, for provision of hold-up capacity while optimum methods are evaluated for treatment, further storage and disposal. The third option is chemical treatment (commonly known as reprocessing or “closed fuel cycle”) in which the various components of the SNF are separated and can be stored/disposed of individually - some may be re-used while others can be converted into more stable waste forms.

Technical issues associated with storage and disposal of SNF include the physical and chemical forms of the waste, waste package design, storage/disposal facility design, safe management, storage capacity and national self-sufficiency. However, non technical issues, such as political acceptance, non-proliferation, and environmental concerns are equally important.

Although no country has yet built a disposal facility for SNF, there is general endorsement of the concept of geological disposal as the end point for spent fuel. Among others, Sweden, Finland, Switzerland and the UK have chosen this management method. In the Swedish decision, which involves direct disposal of the spent fuel, the fuel rods will be packaged into canisters of cast iron fitted with a copper sheath. These canisters will then be placed in the repository, which will be back-filled with clay. The use of these various barriers is to ensure that groundwater does not come into contact with the SNF until sufficient time has passed. This is because reaction between the water and the fuel could conceivably result in release of radioactive materials to the environment. However, in this model, the fuel itself acts as the first barrier to retain radioactive material: the fuel is in the form of uranium dioxide (UO_2), a ceramic material that is very stable and almost unreactive towards water. Further, the fuel itself is clad in Zircaloy, an alloy which is also stable against water and thus forms the second barrier.

In the case of countries that chemically treat fuel, the most highly radioactive wastes are incorporated into glass (by a process known as vitrification), which is very stable against water attack. The vitrified waste is then packed in canisters analogous to those used for the untreated rods in direct disposal.

SNF in Norway has arisen from irradiation of fuel in the NORA, Jeep I and Jeep II reactors at Kjeller, and in the Heavy Boiling Water Reactor (HBWR) in Halden. Some of this fuel is similar to commercial fuel (UO_2 clad in Zircaloy) and may be suitable for direct disposal on the Swedish model or in other repository designs. However, other fuels are chemically reactive and there would be risks associated with direct disposal if they came into contact with groundwater, due to the instability of the material and/or the production of gaseous and potentially explosive reaction products. These materials are fuel manufactured from metallic uranium (U) and fuels clad in aluminium (Al).

Two committees were established in January 2009 to make recommendations for the interim storage and final disposal of spent fuel in Norway. The Phase-2 committee has the mandate to find the most suitable technical solution and localisation for intermediate storage for spent nuclear fuel and long-lived waste. The committee will evaluate different solutions and their associated costs, and then recommend one of the options. The committee will base its work on the work of previous committees and on the report of the Technical Committee (see next paragraph).

Co-incident with the formation of the Phase 2 Committee, the Technical Committee on Storage and Disposal of Metallic Uranium Fuel and Al-clad Fuels was formed. This committee has the mandate to recommend treatment (i.e. conditioning) options for metallic uranium fuel and aluminium-clad fuel to render them stable for long term storage and disposal. The work of the Technical Committee is recorded in this report.

The report presents the current inventory of spent metallic uranium and aluminium fuel in Norway, and continues with a summary of technical information on the behaviour of the fuels relevant to SNF storage and disposal. Accounts of how such materials have been, or are planned to be, treated and stored/disposed of in other countries are included. Based on the scientific background and international experience, the committee's recommendations for storage and disposal of the Norwegian materials are given.

2.2 Summary of the recommendations made by previous Norwegian committees

Work on defining methods for final storage of Norwegian SNF commenced in December 1999, with the establishment of a committee by Royal Decree [NOU, 2001]. This committee, under the leadership of Professor Pål Bergan (the Bergans committee), was given a mandate to develop a national strategy for disposal of high active, spent nuclear fuel.

The committee reported in December 2001. Its report included detailed background information on radioactivity, radiation, the effects of ionising radiation, and nuclear fuels. The main conclusions with respect to the disposal of SNF were:

- The decision on a final disposal method should be delayed while awaiting technical developments in other countries.
- Immediate work should be started on an intermediate storage facility, in which the SNF should be stored for 50 – 100 years.

Other points and/or recommendations that the present committee consider to be relevant were:

- Reprocessing was not investigated as a solution due to the Norwegian Government's objection to the process on environmental grounds.
- Reference was made to the synthetic rock (SYNROC) process in Australia for processing of metallic U fuel, even though a necessary prerequisite to this process is reprocessing.
- It was noted that metallic U fuel must be stored dry because of corrosion problems.
- Investigation of methods of changing metallic U fuel to more stable chemical forms was recommended.
- Norway could consider an international storage / disposal facility, but must then accept the possibility of being the host nation.

Following on from the Bergans committee, a Phase 1 committee was established in 2004, under the leadership of Evelyn Foshaug (Institutt for energiteknikk), with a mandate to:

- Define Norway's requirements for intermediate storage of HLW.
- Investigate possible technical solutions.
- Identify critical points with regard to choice of technical solution and location.
- Suggest a mandate for a future Phase 2 committee.

This committee sat for a relatively short period of time, and reported six months after its establishment [Foshaug et al, 2004]. As technical options for intermediate storage, it recommended further investigation of dry storage in a concrete structure or transportable storage containers. It further recommended that the future Phase 2 committee should choose the final technical solution and location of the storage facility. It also specified the need for a technical committee to investigate safe storage of metallic fuel and fuel with Al cladding.

The committee reported that, within its framework, it was not possible to discuss the option of reprocessing and the associated consequences with regard to the building of an interim storage facility. It was noted that the issue was relevant because of the reactivity of metallic uranium, and because there existed no current international method suitable for storage of metallic fuel. A decision in Norway to reprocess metallic uranium fuel would have a large impact on the dimensions of a storage facility.

2.3 The Phase 2 Committee and the Technical Committee

2.3.1 The Phase 2 Committee

The Phase 2 committee was established by Royal Decree of January 14, 2009, with Professor Erling Stranden appointed as Chairman. Its mandate is to find the most suitable technical solution and localisation for intermediate storage for spent nuclear fuel and long-lived waste. The committee will evaluate different solutions and their associated costs, and then recommend one of the options. The following options will be investigated and assessed against each other:

- Modification of existing facilities in Halden and/or at Kjeller.
- A new storage facility in Halden and/or at Kjeller.
- A new storage facility for spent fuel and long-lived intermediate level waste.

The committee will also assess the competence needed in the organisation and how to maintain this competence, and will also consider the need to decommission the intermediate storage facility, based on IAEA guidelines. The committee will base its work on the Phase 1 Committee report and on the report of the Technical Committee (see next sub section).

2.3.2 Appointment of the Technical Committee on Storage and Disposal of Metallic Uranium Fuel and Al-clad Fuels

Co-incident with the formation of the Phase 2 Committee, the Technical Committee on Storage and Disposal of Metallic Uranium Fuel and Al-clad Fuels was appointed by the Ministry of Trade and Industry (*Nærings- og handelsdepartementet*). The members of the committee were:

Peter Bennett (Chairman)	Chemistry Manager	Institutt for energiteknikk Halden, Norway
Barbara C. Oberländer	Division Head, Nuclear Materials Technology	Institutt for energiteknikk Kjeller, Norway
Pablo Adelfang	Leader, Research Reactors Group, Division of Nuclear Fuel Cycle and Waste Technology	International Atomic Energy Agency Vienna, Austria
Ella Ekeroth	Project Manager	Studsvik Nuclear AB Nyköping, SWEDEN

Ella Ekeroth retired from the committee after six months, and was replaced by Evert Eriksson, Project Manager, also of Studsvik Nuclear. To ensure continuity, Evert Eriksson attended all the committee’s meetings.

Rolf Jullum, Statsbygg (a member of both the Phase 1 and Phase 2 Committees) attended the committee’s meetings as observer. Per Ivar Wethe, Institutt for energiteknikk, Kjeller, provided secretarial and administrative support to the committee.

2.3.3 The Technical Committee's mandate

The detailed mandate of the Technical committee is given in Appendices 1 (original, in Norwegian) and 2 (English translation). In summary, the committee should:

- Propose methods to condition unstable metallic U fuel and/or aluminium-clad fuels to render them eligible for interim storage and final disposal.
- Propose methods for interim storage and final disposal.

Further, the chairman of the Phase 2 Committee communicated to the Technical Committee a wish that, following any conditioning of the fuel to render it suitable for storage, there would be no need for any further treatment or repacking before it can be interred for final disposal [Stranden, 2009].

2.3.4 The committee's interpretation of the mandate

Based on the mandates of the two committees, the requirement for a single conditioning step, and on its perceived requirements for the definition of an end-point for SNF, the Technical Committee has made the following interpretations:

1. The committee should present opinions on all technically feasible options available, without excluding or failing to evaluate any option that might be impossible because of current national laws or known or perceived policy at the time of writing. The exception to this rule was technical options that entail use of UK facilities on the Sellafield site, as these were specifically excluded in the mandate.
2. Any proposed conditioning process should render the fuel in a form eligible for both interim storage and final disposal – or in so-called "road ready" form.
3. Since no decision has yet been taken on the final disposal method for spent fuel in Norway, it has been necessary for the Technical Committee to make some assumptions regarding the final disposition of the fuel elements under consideration. As reported in 2006 by the OECD Nuclear Energy Agency [NEA, 2006], there is general endorsement of the concept of geological disposal. All countries that have made a policy decision on a final step for the management of long-lived radioactive waste (and spent fuel if it is declared as a waste) have selected geological disposal as the endpoint. The committee has thus assumed geological disposal as the basis for its work; more information on this topic is given in Chapter 5.1.

2.3.5 Summary of the work of the Technical Committee

The committee held four one-and-a-half day meetings, in Kjeller, Halden, Studsvik and Vienna. The Kjeller and Halden meetings included visits to the spent fuel storage facilities and discussions with responsible staff. At the Studsvik meeting, the committee received presentations on corrosion testing of glass waste forms and on Studsvik experience with handling and treatment of R1 reactor fuel.

In addition to the full committee meetings, two members (Adelfang and Bennett) attended the IAEA Regional Workshop on the Russian Research Reactor Fuel Return Programme; Lessons Learned, 22 to 25 June 2009, Varna, Bulgaria. Three members (Adelfang, Bennett and Oberländer) and the secretary (Wethe) attended the IAEA Technical Meeting on Guidelines of Good Practice for the Management and Storage of Research Reactor Spent Fuel, 19 to 22 October 2009, Thurso, UK. A paper describing storage of SNF in Norway was prepared for this meeting [Oberländer et al, 2009]. The chairman and secretary of the committee met with representatives of Sosny Company in Halden and Kjeller, and with the Senior General Manager of the Recycling Business Unit of AREVA NC in France.

2.4 Report outline

Chapter 3 presents the current inventory of spent metallic uranium and aluminium-clad fuel in Norway, together with information on irradiation periods, achieved burn-up levels and current storage.

The report continues (**Chapter 4**) with a summary of technical information on the behaviour of metallic uranium and aluminium-clad fuels relevant to SNF storage and disposal.

Chapter 5 presents a general summary of the options for the management of spent fuel, while accounts of how metallic uranium and/or Al-clad fuels have been, or are planned to be, treated and stored/disposed of in other countries are included in **Chapter 6**. This chapter also describes the reasons for the rejection of certain options in some countries.

Options for spent fuel and conditioning products storage, together with accounts of international practice in interim storage of research reactor fuels are described in **Chapter 7**.

In **Chapter 8,** a set of criteria are introduced against which SNF management options can be assessed, and the scientific background and international experience are used to define a list of options for the conditioning of Norwegian metallic uranium and aluminium-clad fuels. Each option is then evaluated in detail. A short discussion of storage options is also included.

Based on the discussion, the committee's recommendations for storage and disposal of the Norwegian metallic uranium and aluminium-clad fuels are listed in **Chapter 9**. Concluding remarks and suggestions for the next steps are given in **Chapter 10**.

3 INVENTORY AND USAGE OF URANIUM METAL FUEL AND ALUMINIUM CLAD FUELS IN NORWEGIAN RESEARCH REACTORS

3.1 Source and irradiation history

The spent nuclear fuel under consideration in this report has arisen from three heavy water research reactors: JEEP I, which was operated at Kjeller from 1951 to 1967, JEEP II, also located at Kjeller, and in operation from 1966 to the present, and the Halden Boiling Heavy Water Reactor (HBWR) in Halden, which commenced operation in 1959 and is still in use. Another reactor, NORA, operated at Kjeller from 1961 to 1968. The fuel, which was identical to that used in JEEP I, was returned to the USA, and thus is not discussed further in this report.

Some spent fuel from JEEP I was used in a pilot reprocessing plant at the Kjeller site, which was in operation from 1961 to 1968, and later decommissioned. The second core loading of the HBWR was reprocessed in Belgium in 1969. The recovered uranium and plutonium was sold for civilian use, and the waste was disposed of in Belgium. With these exceptions, all Norwegian spent fuel is stored at Kjeller and in Halden.

In total Norway has some 16 tonnes of spent fuel, of which six tonnes are stored at Kjeller and 10 tonnes in Halden. There is approximately 12 tonnes of aluminium-clad fuel, of which 10 tonnes is metallic uranium fuel and the remainder oxide (UO_2). The committee’s mandate encompasses all of the JEEP I and JEEP II fuels and the first loading of the HBWR (subsequent loadings of the Halden reactor used oxide fuel clad in Zircaloy), which amounts to 75 per cent of the total SNF inventory.

Table 3-1 lists, where available, information on the source of the fuel material and the producer of the fuel rods. This information is necessary when, for example, investigations are made into whether the fuel can be returned to the country of origin (Chapter 6).

Some details of the fuel rods (fuel and cladding material, dimensions, amounts and burn-up levels) are given in Table 3-2. In summary, the JEEP I and the HBWR 1st charge fuels were metallic uranium clad in aluminium, while the JEEP II fuel is 3.5% enriched UO_2 clad in aluminium. Figures 3-1 to 3-3 show drawings of JEEP I, JEEP II and HBWR 1st charge fuel rods.

The JEEP 1 fuel was, as mentioned above, all discharged by 1967, with typical burn-ups in the range from 200 to 400 MWd/t U.

The Halden reactor first charge fuel was irradiated in the period from 1959 to 1962, for 1000 hours at low temperature (approximately 150°C) and low power (reactor power 6 MW). The total power generated in the period was approximately 79 MWd, equivalent to 0.27 MWd per element and an average rod burn-up of approximately 12 MWd/t U. The irradiation conditions were such that all the produced plutonium was in the form of the ^{239}Pu isotope.

On average, JEEP II fuel elements are irradiated for 10 years, to a burn-up of 15000 - 16000 MWd/t U.

There is currently no metallic uranium fuel in either the HBWR or JEEP II, and thus no further spent fuel of this type will be produced. The JEEP II core contains about 220 kg of fuel (UO_2 clad in aluminium), and approximately 45 kg per year is discharged.

Table 3-1. Fuel material source and fuel rod producer

Reactor Fuel type	**Fuel**	**Enrichment done / delivery of fuel powder**	**Producer**
JEEP I			
Standard element / void element	Metallic natural U	Not applicable – not enriched	United Kingdom France Netherlands Norway
Seed elements	1.7% UO_2	USA	IFE, Norway
HBWR			
1st charge	Metallic natural U	Not applicable – not enriched	United Kingdom
JEEP II			
1st charge	3.5% UO_2		IFE, Norway
2nd charge	3.5% UO_2		IFE, Norway
3rd charge	3.5% UO_2		IFE, Norway
4th charge	3.5% UO_2		IFE, Norway
5th charge	3.5% UO_2		IFE, Norway
6th charge	3.5% UO_2	Germany	IFE, Norway
7th charge	3.5% UO_2	Sweden	IFE, Norway

Table 3-2. Details of fuel rods

	HBWR	JEEP I	JEEP I Seed	JEEP I Void	JEEP II
Fuel material	Metallic uranium	Metallic uranium	UO_2	Metallic uranium	UO_2
Cladding material	Aluminium	Aluminium	Aluminium	Aluminium	Aluminium
Total mass U, tons	6.7	3.1	75 kg	37 kg	2.0
Number of fuel rods	308	176	60	7	1050
Fuel rod diameter, cm	2.54	2.54	1.27	2.54	1.28
Fuel length, cm	237.5	190 (active fuel stack)	96.5 (active fuel stack)	190	90
Initial enrichment, %	Natural uranium	Natural uranium	1.7	Natural uranium	3.5
Irradiation period	1000 hours (1959 – 1962)	1951 - 1967	1951 – 1967	1951 - 1967	1966 to present
Burn-up, MWd/t U	12	1 – 1000 (mostly 200 – 400)	570-1000	NA	Average 15 000

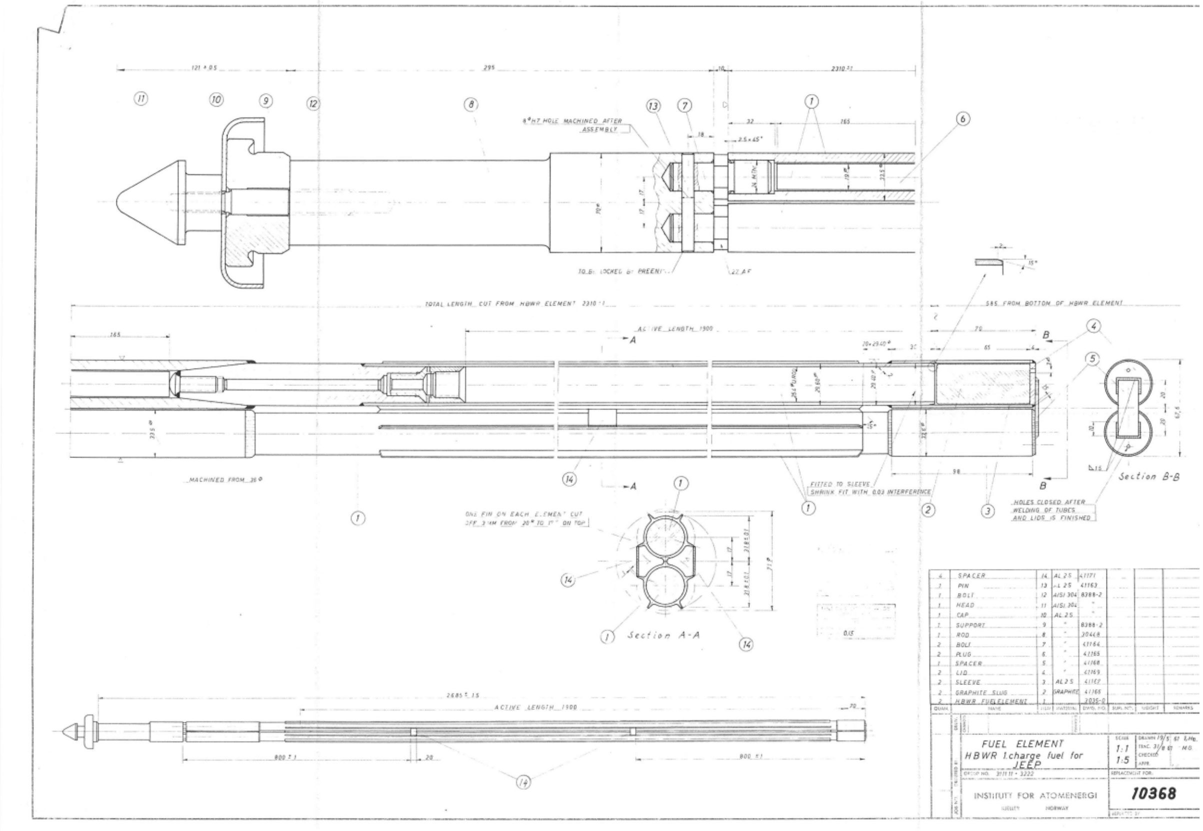


Figure 3-1. Drawing of JEEP I fuel

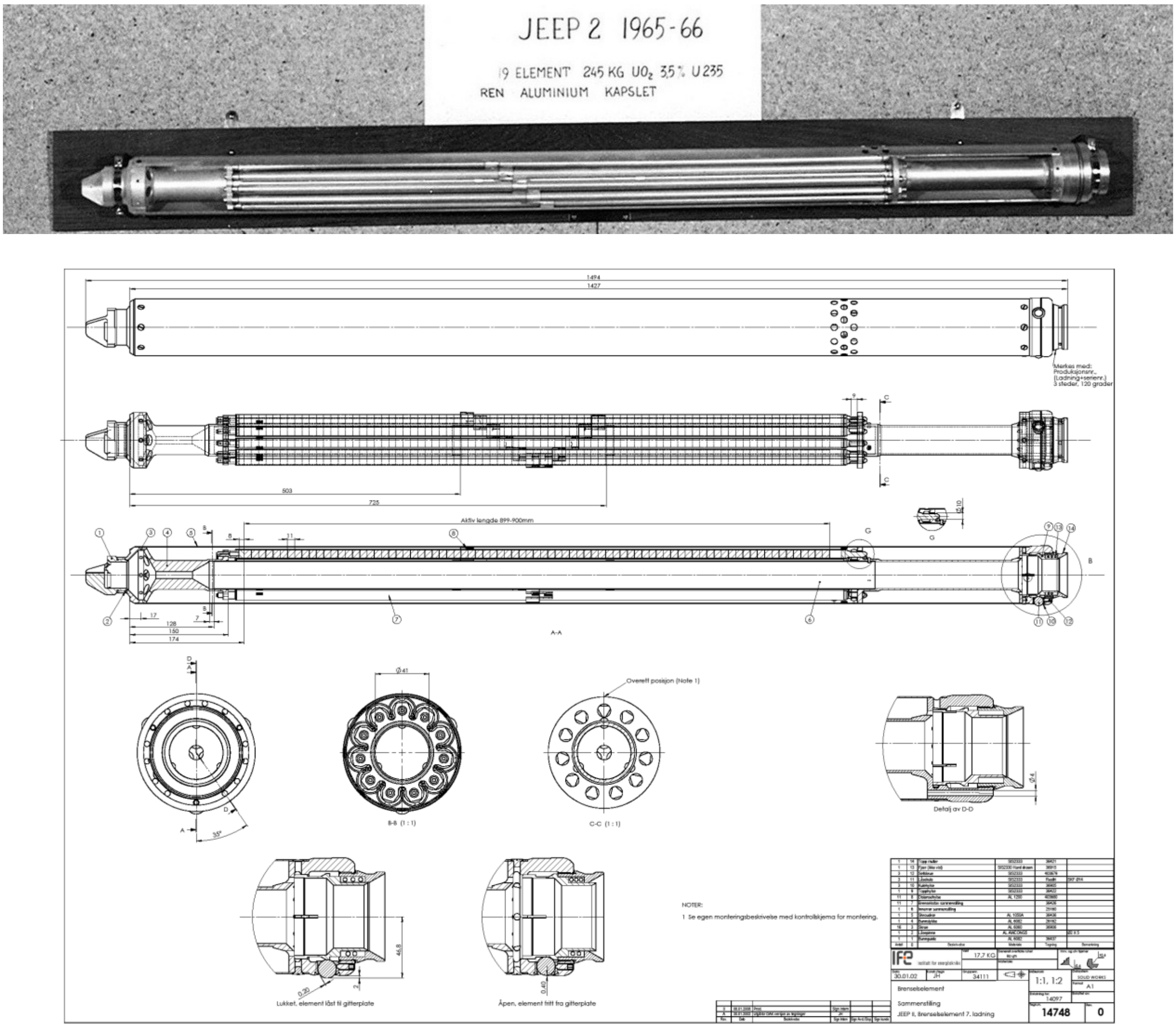


Figure 3-2. Drawing of JEEP II fuel

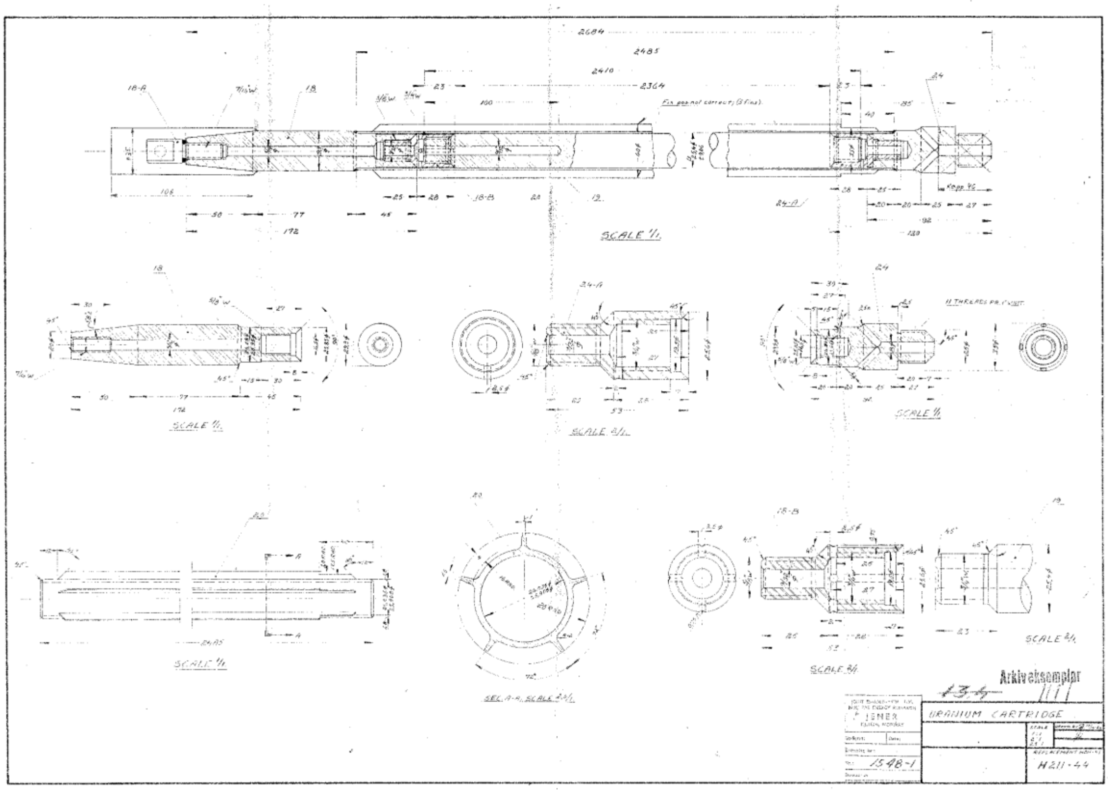


Figure 3-3. Drawing of HBWR 1st charge fuel

3.2 Spent fuel management

The spent fuel from the three reactors is stored separately in dry storage facilities: the JEEP I fuel in the "JEEP I Stavbrønn" at Kjeller, the JEEP II fuel in the "Met. Lab II" building at Kjeller, and the first charge HBWR fuel in the "Bunker Building" in Halden. These facilities are described in the Bergans and Phase 1 committees reports [Bergans, 2001, Foshaug, 2004]; a brief recap together with some additional information is given below.

3.2.1 JEEP I fuel

In the early days of JEEP I operation, fuel assemblies were subjected to interim inspections during reactor outages. The fuel was discharged if signs of cladding degradation were observed, such as blisters, spallation of the surface oxide, or cracks.

After discharge, JEEP I fuel was initially stored in aluminium baskets in a wet storage facility, which included a water purification system. The maximum wet storage period was 10 years. Transfer of the fuel to the JEEP I dry storage facility started in 1961, and the final transfer took place in 2000 (from the "Bunker Building").

The JEEP I dry storage facility consists of top and bottom concrete slabs, with the top one at ground level and the bottom approximately 3 m underground. There are 97 vertical holes in each block, and in each hole is positioned a 2.7 m long stainless steel pipe, which is fixed into the 20-cm thick slab. The top of each tube is covered with a metal cap and an expansion gasket. The space around the tubes between the two slabs is filled mainly by sand. The storage block is covered by a free-standing building (Figure 3-4).

Figure 3-4. JEEP I dry storage building

The fuel elements were inspected and transferred/repacked from Al baskets into AISI 304 stainless steel baskets in 1982. Actions were taken to dry moisture/condensed water in the outer stainless steel tubes and to reorganise the inventory. One inner steel basket is stored in each of the outer stainless steel tubes; each basket contains either one or two spent fuel assemblies (SFA; each comprising two fuel rods).

An inspection of the fuel was performed in 1982 when it was re-packaged. Some cladding defects were observed, such as oxide spallation, ballooning, blisters, pittings and cracks. Most of the defects were present in early fuel, discharged in 1951. For each SFA, a log-card was produced, containing dates on fuel handling and storage, data on burn-up, and remarks from visual inspections of the cladding. Table 3-3 summarises the findings of the 1982 inspection.

Table 3-3. Observations from 1982 inspection of JEEP I fuel

Observation	**Oxide spallation**	**Balloon**	**Blisters**	**Pittings**	**Cracks**
No of affected SFA	2	7	27	5	31
Fuel discharge date	1951/65	1951	1951	1951	1951

3.2.2 JEEP II fuel

Spent fuel from JEEP II is stored in water-filled wells (with a purification system) for six to 12 months before being transferred to the dry storage facility in the METLAB II building (Figure 3-5). This consists of a concrete block, with its top at ground level. The block houses 84 vertical steel tubes, 32 of which have a depth of 3 m, while the remaining 52 are 3.5 m deep. All the tubes are sealed by lead plugs. The concrete block drains through a pipe to a delay tank. The concrete block is located beneath a building specifically designated for loading and unloading of transports of radioactive material.

Figure 3-5. JEEP II fuel storage facility

3.2.3 First charge HBWR fuel

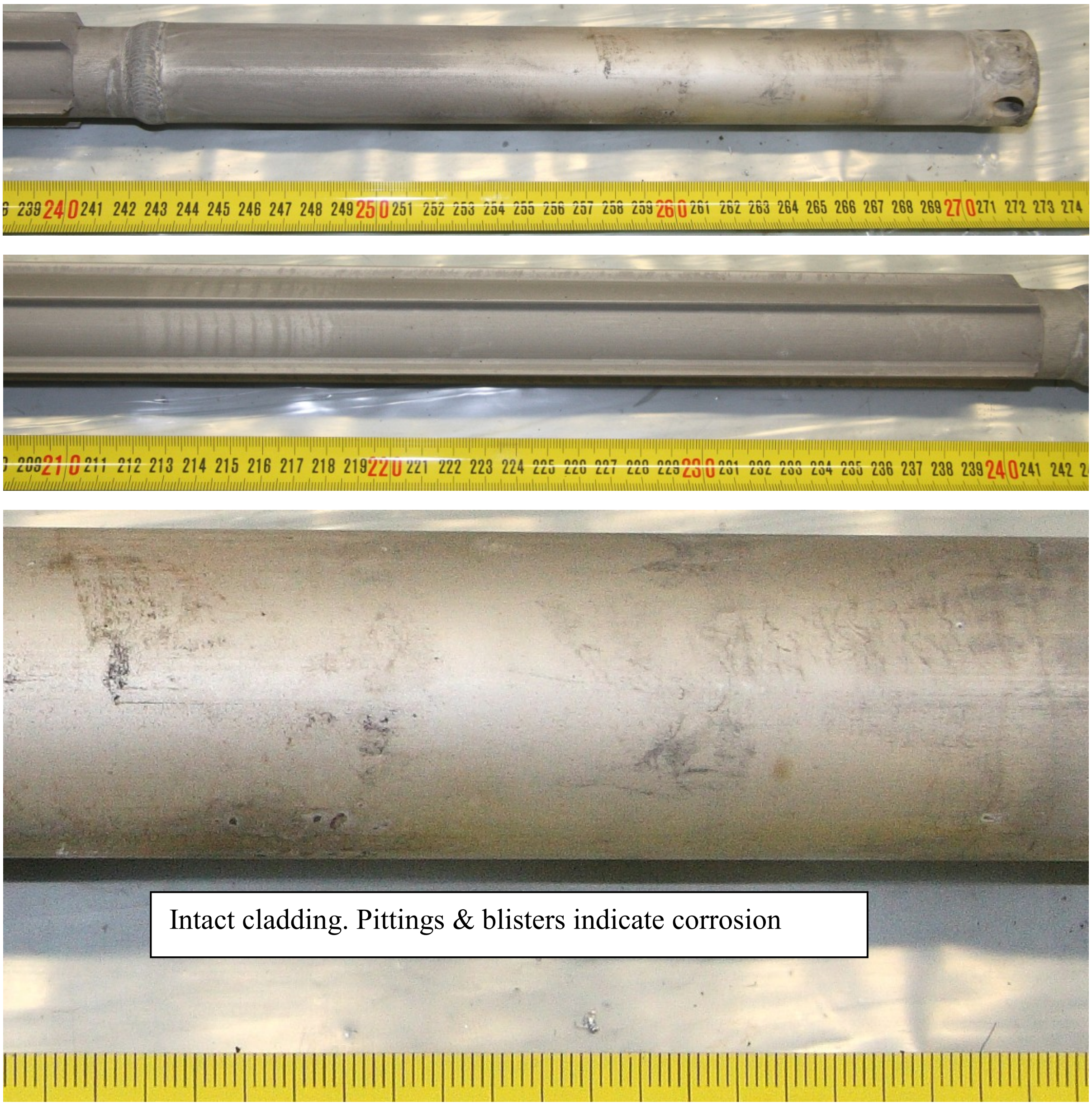
The period of wet storage for the 1st charge HBWR fuel was less than three years. In 1962, the rods were dried and loaded into aluminium storage capsules. Inspection showed that all the fuel rods were intact. These capsules were loaded into the dry storage facility.

The storage facility consists of a concrete construction with 2 m thick walls, with a 1 m thick front shield of reinforced concrete. There are 202 holes in the front, in which 7 m long steel tubes are fitted horizontally. A metal frame to support the tubes is fitted inside the concrete construction. The steel tubes are cooled by natural circulation of air. This facility, together with a wet storage pond, is positioned inside a building fitted with a ventilation system (Figure 3-6).

The facility was sealed (by the IAEA) until 2006, and thus no inspections were done. Subsequent observations showed that the storage tubes were dry.

Figure 3-6. HBWR Bunker Building and dry storage facility

In connection with the work of the committee, a rod was selected randomly for visual inspection. After 47 years in dry storage, the cladding was intact, although signs of incipient corrosion were observed (Figure 3-7).



Intact cladding. Pittings & blisters indicate corrosion

Figure 3-7. Visual inspection of HBWR 1st charge fuel in 2009

4 CHEMICAL AND PHYSICAL PROPERTIES OF METALLIC URANIUM FUEL AND ALUMINIUM CLADDING RELEVANT TO SPENT FUEL STORAGE AND DISPOSAL

4.1 Metallic uranium

The following discussion of the reaction between metallic uranium and water is based primarily on two reports. A recent report issued by Pacific Northwest National Laboratory (PNNL) for the United States Department of Energy [Delegard and Schmidt, 2008] contained a comprehensive review of the technical literature as well as reporting the results of experiments carried out at PNNL. In addition, the review of United States Department of Energy (USDOE) spent fuel corrosion of Hilton has been used [Hilton, 2004].

The reaction between metallic uranium and oxygen-free (anoxic) water forms uranium dioxide (UO_2) and hydrogen gas (H_2), as described by the following reaction:

$$U + 2H_2O \rightarrow UO_2 + 2H_2$$

The reaction is highly exothermic: i.e. it releases energy to the surroundings. The enthalpy of reaction, ΔH_{298}, is -513 kcal/mol of uranium, which is nearly twice that of the reaction between hydrogen and oxygen gas to form water.

Once a small amount of hydrogen is generated, it can react directly with uranium to form uranium hydride (UH_3):

$$2U + 3H_2 \rightarrow 2UH_3$$

Because of the high diffusivity of H_2, uranium hydride is often found along cracks, grain boundaries or in crevices, such as beneath cladding. Pure uranium hydride is pyrophoric (i.e. it ignites spontaneously in air) [Katz, Seaborg and Morss, 1986]

The UH_3 can then function as an intermediate by its reaction with water that forms further uranium dioxide and hydrogen gas:

$$2UH_3 + 4H_2O \rightarrow 2UO_2 + 7H_2$$

The hydrogen produced in this reaction can react with more uranium metal, or can escape into solution or the gas phase. If the hydrogen cannot escape, then the system pressure may be raised.

Further oxidation of UO_2 can occur due to the presence of oxygen, to form the following products:

- Uranites: UO_2, U_4O_9, U_3O_7.
- Triuranium octaoxide: U_3O_8.
- Schoepites and uranyl peroxides.

The uranium oxide that is formed by reaction of uranium with water is not protective and spalls from the surface. When the oxide product thickness reaches about 0.1 µm it spalls off as sheets.

The reaction occurs via a three.-stage mechanism (Figure 4-1):

1. Adsorption of oxygen molecules.
2. Dissociation of adsorbed oxygen molecule.
3. Continued grown by diffusion or migration of oxygen and metal ions through oxide.

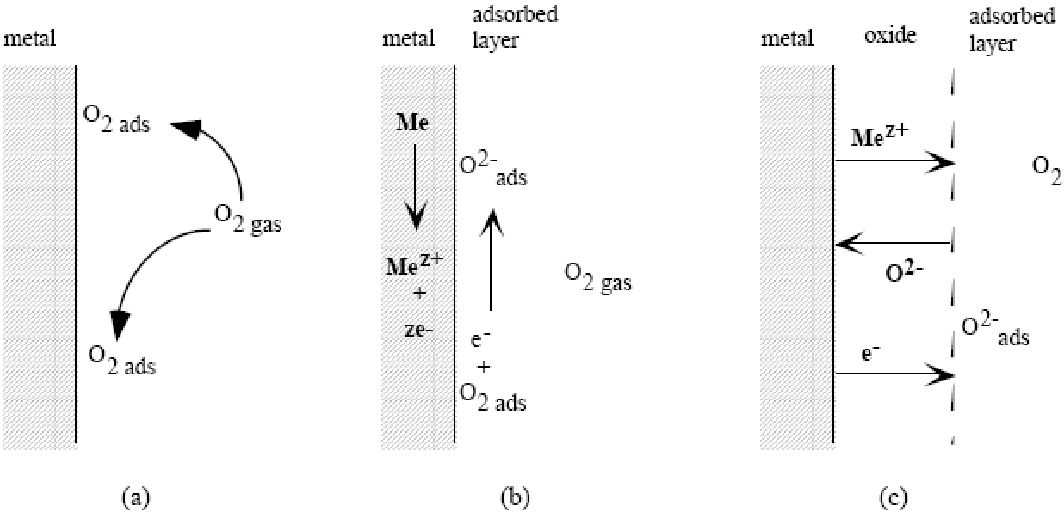


Figure 4-1. Reaction mechanism of metallic uranium with water

The observed paralinear oxidation rate can be described by an inner coherent layer that transforms to an outer porous oxide at a linear rate resulting in a constant thickness layer. The oxidizing species (i.e. water) penetrate the outer porous layer at a rate determined the diffusion through the inner oxide layer to the metal-oxide interface.

The corrosion rate of uranium metal is initially parabolic but becomes linear (i.e. proportional to time) following cyclic spallation and re-generation of the non-adherent uranium oxide layer.

Uranium metal corrosion in anoxic water is generally uniform; i.e. the corrosion takes place uniformly over the entire exposed surface.

The rate of the reaction between uranium and water has been measured by the following methods:

- Weight loss of U metal coupon after removing the UO_2 layer.
- Weight gain of the U based on formation of UO_2.
- Decrease in size of U metal coupon.
- Quantity of product hydrogen gas.
- Quantity of released fission product gas (Kr, Xe) from irradiated metal.
- Quantity of released fission product ^{137}Cs from irradiated metal.
- Electrochemical (current/potential) measurements of corrosion.

The PNNL study reviewed 32 studies containing 128 data points, and showed remarkable consistency between the results. The uranium metal corrosion reaction rate follows an Arrhenius equation in the temperature range from ~ 24°C to 350°C, as shown graphically in Figure 4-2:

$$k = B\ \exp(-E_a/RT)$$

where: k = rate

B = pre-exponential factor

E_a = Activation energy (kcal/mol or kJ/mol)

R = gas constant, 8.314 (J/(mol K))

T = absolute temperature (K)

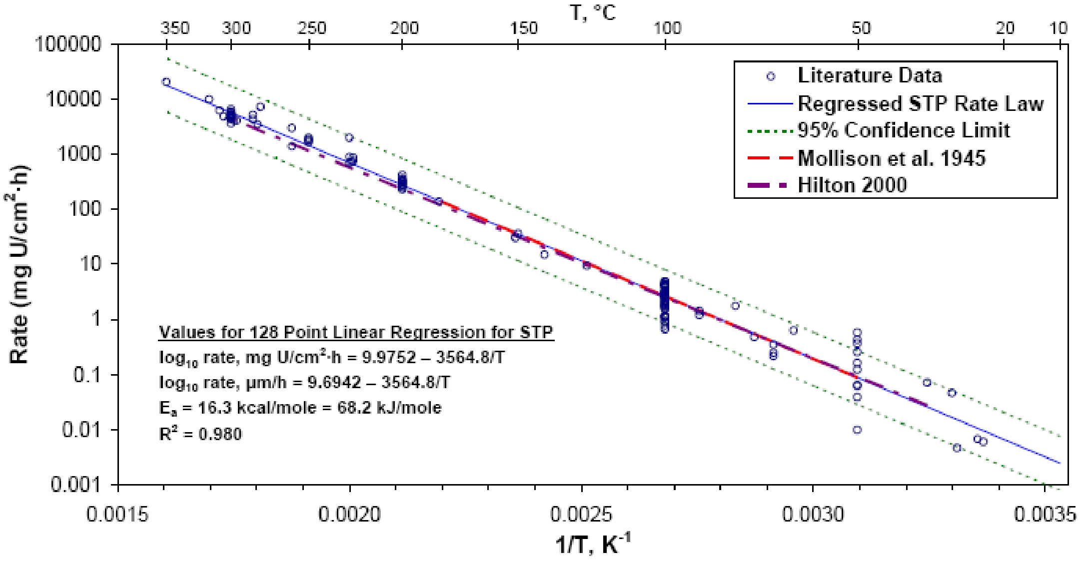


Figure 4-2. Reaction rate of uranium metal with water

Mathematical re-arrangement gives:

$$\ln k = -E_a/RT + \ln B$$

$$\log_{10} k = -E_a/(2.303\ RT) + \log B = A/T + B$$

where $A = -E_a/(2.303\ R)$

This equation can be written quantitatively in different ways, giving the reaction rate in terms of per unit surface area of the sample or depth of linear corrosion attack:

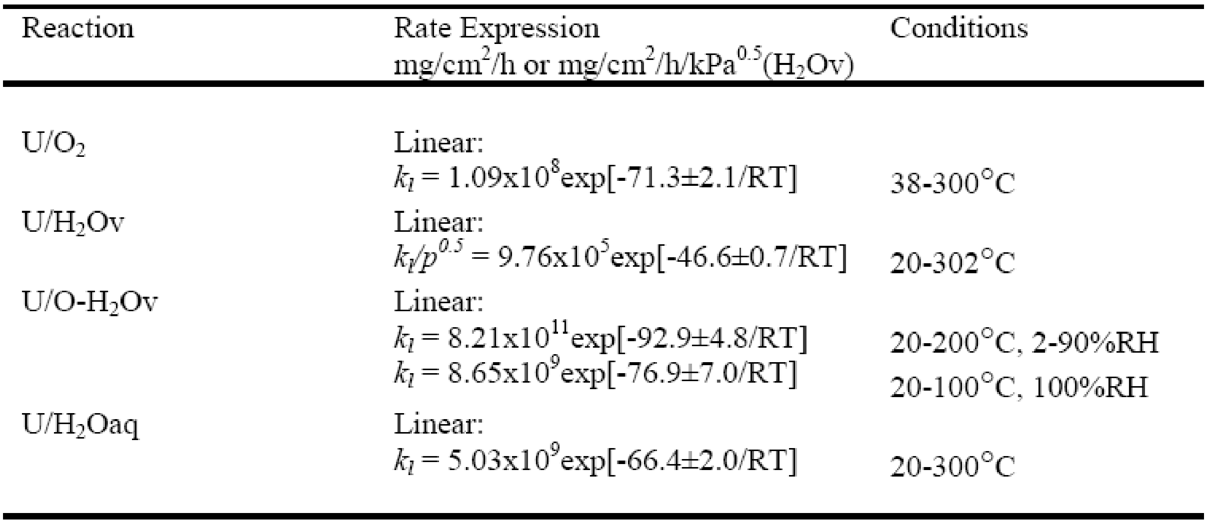
$$\log_{10} k\ (\text{mg U/cm}^2\ \text{h}) = 9.9752 - 3564.8/T$$

$$\log_{10} k\ (\mu\text{m/h}) = 9.6942 - 3564.8/T$$

The reaction between metallic uranium and oxygen gas (O_2) forms UO_2 according to the reaction:

$$U + O_2 \rightarrow UO_2$$

The reaction rate of the U/O_2 reaction is around three orders of magnitude lower than the water driven reactions. Above a threshold oxygen concentration the oxygen inhibits the water driven rates to values intermediate to the dry oxygen and anoxic water rates. The rates of reaction for uranium with oxygen, water vapour, oxygenated water vapour and water at temperatures less than 300 °C (and p < 101 kPa) have been summarised by [Hilton, 2000]:



Reaction	Rate Expression $mg/cm^2/h$ or $mg/cm^2/h/kPa^{0.5}(H_2Ov)$	Conditions
U/O_2	Linear: $k_l = 1.09x10^8 exp[-71.3\pm2.1/RT]$	38-300°C
U/H_2Ov	Linear: $k_l/p^{0.5} = 9.76x10^5 exp[-46.6\pm0.7/RT]$	20-302°C
$U/O\text{-}H_2Ov$	Linear: $k_l = 8.21x10^{11} exp[-92.9\pm4.8/RT]$ $k_l = 8.65x10^9 exp[-76.9\pm7.0/RT]$	20-200°C, 2-90%RH 20-100°C, 100%RH
U/H_2Oaq	Linear: $k_l = 5.03x10^9 exp[-66.4\pm2.0/RT]$	20-300°C

Studies have shown that irradiation does not increase the intrinsic oxidation rate of uranium metal, although the apparent rate (normalized to geometric surface area) increases with irradiation induced swelling. The oxidation rate of metal fuel irradiated to low burnup (less than 3000 MWd/MTU, with negligible swelling) is expected to be similar to that of unirradiated U metal.

Metallic uranium is pyrophoric, i.e. it is liable to explode [Epstein et al, 2000]. The presence of pyrophoric uranium metal and uranium hydride, together with that of hydrogen gas, has resulted in several incidents with metallic uranium fuel. Decades ago, uranium fires were commonplace at US Atomic Energy Commission facilities. A review of the literature on this subject [Epstein et al, 2000] was conducted recently in connection with the Hanford site remediation in the USA, as these phenomena are important to, among others, processing of spent nuclear fuel.

The factors which influence uranium pyrophoricity are metal particle size, ambient temperature, ambient moisture content, and heat sources other than oxidation-kinetic heating. Uranium metal ignition can be categorized into the following four types:

1. Onset (ignition) of a chemical runaway reaction inside opened drums containing clad uranium fuel elements with exposed surfaces (due to corrosion or sectioning). High ambient temperature and humidity were presumed to be responsible for these ignitions. Thousands of spontaneous fires have been experienced at room temperature. Another common experience is

the ignition of porous or powdered uranium metal or metal chips in air after resting uneventfully for several days in an open container.

2. Explosions inside drums containing corroded fuel elements or metal scrap. An explosion occurred while the drum was being tapped to loosen the lid and at the instant that ambient air entered the drum. Hydrogen gas was also present.

Explosions of storage drums have been observed when fuel is found to be severely corroding in their containers. In many reported incidents, flashing of the fuel or explosions occurred when the containers were opened. The flashing was believed to be the spontaneous ignition of uranium or uranium hydride powder which became suspended due to the mechanical disturbance of opening the container. Suspended uranium powder could also be the cause of the explosions. Hydrogen gas is usually present inside containers that store corroded fuel, and this will exacerbate the situation because ignition requirements are reduced for combined flammable gas and dust mixtures in air.

Metal dust is quite flammable and explosive. Depending on the dust concentration, the flame speed may be high and comparable with that in gas deflagrations. The hazard of an uranium dust explosion upon opening a container containing corroded fuel is about the same as a hydrogen explosion.

3. Ignition of highly corroded fuel elements or defueled, highly porous cladding following accidental dropping onto the floor, or during element-to-element contacting under violent shaking conditions in fuel dissolvers, or while in transfer trays.

4. Ignition of badly corroded or powdered uranium metal in air at ambient temperature, or ignition of accumulated metal powder under water.

4.2 Aluminium

This section of the report is based on a review published by the IAEA on the corrosion of research reactor aluminium clad spent fuel in water [IAEA 2003b]. For the purposes of this review, it is assumed that the reactions of aluminium in repository groundwater will be the same as those in spent fuel storage ponds.

A large proportion of research reactor fuel in both eastern and western countries is fabricated with a core consisting of uranium-aluminium alloys and clad in aluminium alloy. Aluminium is used as cladding because of its low absorption of neutrons. In the past, aluminium corrosion was not a problem because the fuel was normally reprocessed. However, since the cessation of reprocessing, some corrosion problems have been observed.

Figures 4-3 and 4-4 show typical corrosion damage exposing fuel meat on RA-3 aluminium fuel assemblies [Vinson et al, 2002]. In the first figure, corrosion nodules can be seen, and each appears to have originated in a scratch on the surface of the fuel clad. Figure 4-4 shows the cladding damage that is apparent with the corrosion product nodules dislodged, exposing the fuel meat.

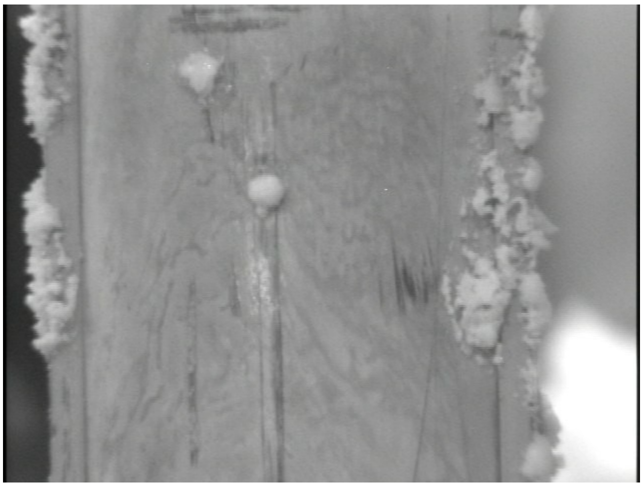


Figure 4-3. Corrosion nodules associated with scratches and crevices on Al cladding

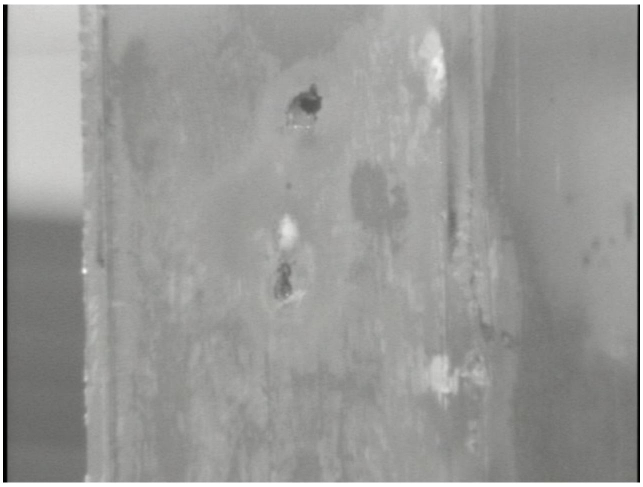


Figure 4-4. Pits through Al cladding

The corrosion of aluminium clad spent nuclear fuel is dependent on a number of inter-related factors. These factors may operate singly or synergistically, making predictions difficult. Many of the metallurgical factors are already inherent in the cladding when it is delivered by the manufacturer. Factors such as alloy composition, heat treatment, microstructure, nature and thickness of the oxide coating, inclusions and impurities in the alloy and cold work all play a role in the corrosion process, but they cannot be controlled during storage or deposition.

Aluminium is one of the most reactive of all the metals, and, paradoxically, it owes its stability to this reactivity since a protective oxide barrier is rapidly formed on its surface, which prevents further corrosion of the metal. However, if the oxide film is damaged under conditions where self healing does not occur, localised corrosion can occur, leading to extensive corrosion.

In most water environments, the overall reaction between aluminium and water forms aluminium hydroxide ($Al(OH)_3$) and hydrogen gas (H_2), as described by the following reaction:

$$2Al + 6H_2O \rightarrow 2Al(OH)_3 + 3H_2$$

Several different modes of Al corrosion can occur. General corrosion, i.e. uniform metal corrosion or metal thinning over the entire surface of an aluminium-clad fuel rod, is not a problem in wet storage, and thus would not lead to cladding failure in a repository. As a general rule, the protective oxide film is very stable in the pH range from 4.0 to 8.5.

Galvanic corrosion occurs when two dissimilar metals are electrically coupled together in an electrolyte (an electrically conducting solution). The corrosion of the less corrosion resistant material increases. Aluminium and its alloys are highly susceptible to failure by galvanic attack. In particular, stainless steel promotes corrosion of aluminium: this is significant as many spent fuel elements are stored in stainless steel containers.

Crevice corrosion is a highly localised form of corrosion and occurs on closely fitted surfaces upon entry of water into the crevice. The mechanism of the reaction is complex, but depends on the transport of chloride ions into the crevice, causing the acidification of the local environment. Metals that depend on oxide films for corrosion resistance, such as aluminium, are particularly susceptible to crevice corrosion. Crevices can occur in many locations in complicated structures. These locations provide the environment for localised corrosion to occur because of stagnant areas.

In wet basin storage, pitting has been the main mechanism of aluminium corrosion. Pitting is a localised form of corrosion in which metal is removed preferentially over very small areas on the surface to develop cavities or pits, which results in holes being produced in the metal. It is one of the most destructive and insidious forms of corrosion. The attack is generally limited to extremely small areas, while the remaining surface is relatively unaffected. The pits usually start at small points on the surface and enlarge with time. Pitting is most common on metals covered with an adherent surface film. The pits tend to develop at defects or flaws in the surface film and at sites where the film has been mechanically damaged and does not self-heal. Under special conditions, the rate of pitting can be rapid and can lead to perforation of the fuel cladding.

The corrosion of aluminium alloys in water is complex, and many of the factors affecting the corrosion are interrelated. In general, soft water is less aggressive than hard water with respect to pitting corrosion of Al. The hardness is due to the presence of calcium carbonate ($CaCO_3$), which is present in many groundwaters.

The major factors believed to influence the pitting of aluminium alloys are conductivity, pH, and bicarbonate (HCO_3^-), chloride (Cl^-), sulphate (SO_4^{2-}) and oxygen content. Because of the interrelationship between the water composition and fuel rod service factors, it is difficult to predict the influence of water on Al corrosion from a table of water composition alone. A number of studies using synthetic waters containing several metal and salt ions alone and in combination have shown that Al corrosion is accelerated when salts of copper, chlorides and bicarbonates were present together, compared with cases where only a single impurity was present.

The protective oxide film on the surface controls the corrosion rate and pitting tendency of aluminium alloys. The corrosiveness of complex waters are due to the abilities of ions to penetrate the oxide film and attack the Al metal. The pitting corrosion of Al is often caused by halide ions, of which chloride is the most common. Chlorides break down the protective oxide film and inhibit repassivation. Chlorides migrate into pits and crevices, causing rapid dissolution. Heavy metal ions such as copper and mercury are very aggressive with respect to pitting corrosion of aluminium alloys. A copper concentration of 0.1 ppm can cause pitting of Al. The bicarbonate ion alone does not cause pitting at concentrations up to 300 – 400 ppm, but causes intensive pitting in the presence of chloride and copper.

At repository depths, groundwaters are inevitably oxygen-free [Shoesmith, 2007]. It has been found that in water containing copper, bicarbonate and chloride ions, the exclusion of air reduced corrosion to one third of that in oxygen saturated water.

Formation of blisters or raised areas in the cladding of spent nuclear fuel can lead to breach of the Al cladding. This blistering is a manifestation of internal gas pressurisation and/or internal oxide formation. Blistering is facilitated under oxides because hydrogen has low diffusion rates in Al, so trapped hydrogen disperses slowly. Blisters can be formed by several mechanisms: (i) during manufacture, by trapped air or inclusions of intermetallic particles; (ii) during irradiation, by agglomeration of fission gas products; (iii) due to corrosion, by hydrogen gas formation.

A large amount of experience of aluminium clad corrosion has been obtained in the USA since the USNRC took the decision in the late 1980s to halt processing of nuclear materials for defence purposes under the non-proliferation policy. At the Savannah River Site (SRS), most of the 200 tonnes of stored fuel was clad with Al. Some of this fuel had been stored for over seven years, and showed signs of extreme corrosion. Similar extended storage pertained at other sites (Hanford, Idaho Technologies Co, Oak Ridge National Laboratory); significant problems existed with the Al-clad fuel at some sites, while at others corrosion was minimal or non detectable.

As a consequence of the USDOE's decision to allow return of foreign research reactor spent fuel to the USA (this program is discussed in more detail in Chapter 6.6), over 1700 Al clad assemblies had been inspected for corrosion and mechanical damage by 2003. The condition of the assemblies ranged from pristine to severely corroded, with nodular corrosion and pitting. Approximately seven per cent of the assemblies showed pits that had breached the cladding. Some Al-clad fuels have been in water storage for more than 40 years and remain in pristine condition, while others are severely degraded by pitting corrosion.

5 OPTIONS FOR SPENT FUEL MANAGEMENT

This chapter presents a general account of the possible options for management of spent nuclear fuel. Most of the options discussed in this chapter are mainly applicable to commercial reactor fuel (UO_2 fuel clad in Zircaloy), but may also be suitable for research reactor fuel. Other options, which have been developed specifically for research reactor fuel, are discussed in Chapter 6, which presents international experience in dealing with metallic uranium and aluminium-clad fuels. It must be noted that not all options are available to all countries. The possibility of using a particular management strategy and/or process depends on the availability of domestic technology and resources or on the willingness or otherwise of another country to share technology or offer commercial services.

A recent IAEA report has stated that, currently, the choice of a strategy for the treatment of spent nuclear fuel (the so-called back-end strategy) is limited to the following three possibilities [IAEA, 2008a]:

1. “Direct disposal” or “once through fuel cycle”;
2. “Storage and postponed decision” or “wait and see option”;
3. “Reprocessing and recycling” or “closed fuel cycle”.

5.1 Direct disposal

Although no country has yet built a disposal facility for SNF, there is general endorsement of the concept of geological disposal as the end point for spent fuel. As reported in 2006 by the OECD Nuclear Energy Agency (NEA) [NEA, 2006]:

> *All countries that have made a policy decision on a final step for the management of long-lived radioactive waste (and spent fuel if it is declared as a waste) have selected geological disposal as the endpoint. (This statement applies worldwide, not only to OECD member countries...).*

Information was based on the national reports on the Joint Convention submitted by IAEA member states to the first review meeting [IAEA, 2003c]. Countries reporting a policy, a preference or a reference option for eventual geological disposal included Australia, Belgium, Canada, Czech Republic, France, Finland, Germany, Hungary, Italy, Japan, Korea, Netherlands, Norway, Sweden, Switzerland, and the USA.

Progress has been made in several countries, in which definite and practical steps have been made towards implementing geological disposal. In France, the 2006 Programme Act on the sustainable management of radioactive materials and wastes retains retrievable deep geological storage as the reference solution for long lived and high level waste [IAEA, 2008a].

On June 3, 2009, the Swedish Nuclear Fuel and Waste Management Company, SKB, announced that it had decided to select Forsmark as the site for the final repository for Sweden’s spent nuclear fuel [SKB, 2009a]. All spent nuclear fuel from Swedish nuclear power plants will be disposed of in the final repository at a depth of nearly 500 metres in the crystalline bedrock. SKB will now proceed to complete applications for permits that will be reviewed by the Swedish Radiation Safety Authority and the Environmental Court. The applications will be submitted in 2010 and include an environmental impact assessment and a safety analysis.

The Finnish utility IVO (which operated Soviet reactors) returned spent fuel to the Soviet Union (Russia) for reprocessing, without return of HLW to Finland, until 1994 when it was prohibited by a change in law [NEI 2009]. The other utility, TVO, which had bought reactors from Sweden, was unable to dispose of its HLW abroad as Western reprocessing facilities were mandated to return HLW. This led to a national program to study geological disposal in Finland. Spent fuel is to be stored for 40 years in an interim storage facility, and is considered then to be suitable for disposal. The selection of the Olkiluoto site was accepted by authorities, local community and government in 2000 and by parliament in 2001.

The disposition method is essentially identical in the two countries, and is based on three protective barriers [SKB, 2009b]. The engineered barriers are the low solubility spent fuel (which is UO_2, as compared with the highly soluble metallic U fuel discussed in this report), a copper-iron canister (where the copper works as a barrier against corrosion and a cast nodular graphite iron insert gives the required mechanical strength), and highly compacted bentonite clay. This clay has the ability to expand when it comes into contact with water so it will limit the movement of groundwater near the canister. A schematic is shown in Figure 5-1.

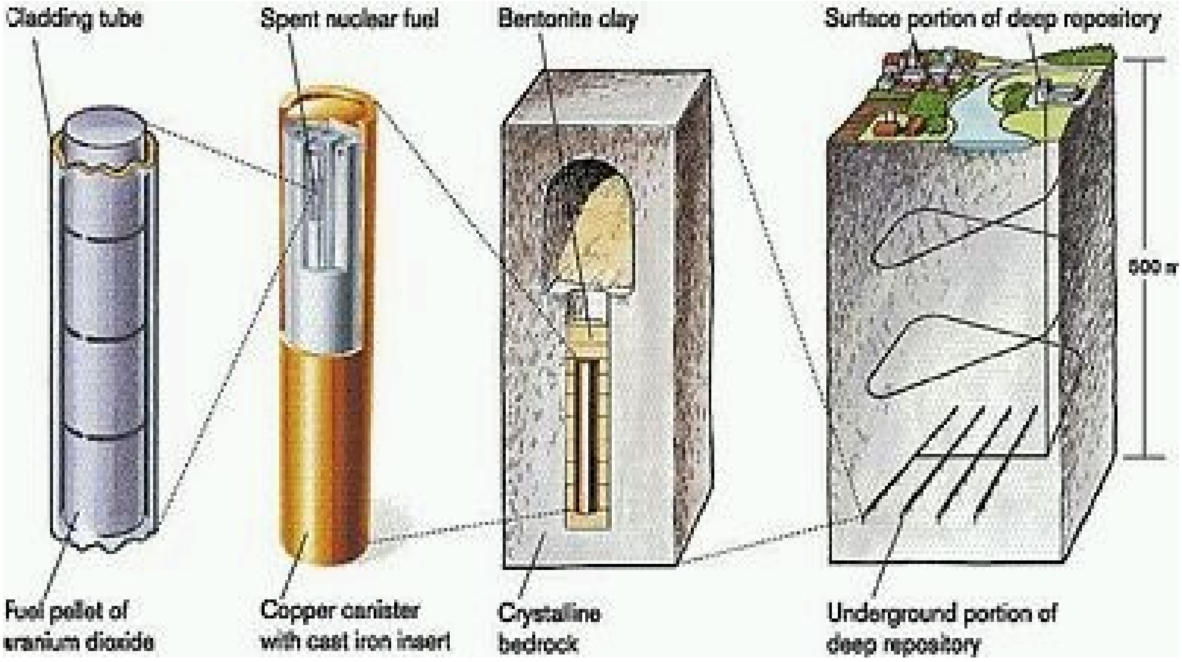


Figure 5-1. Barriers for geological disposal of commercial reactor fuel

The canisters are placed in crystalline basement rock at a depth of about 500 metres, embedded in bentonite clay. After disposal the tunnels and rock caverns are sealed. The impermeable copper canister fully contains the spent fuel. The bentonite buffer protects the canister against corrosive attack and rock movements. If a fracture occurs in a canister, the bentonite clay buffer and undamaged parts of the canister will prevent water from penetrating into the canister. The buffer will also prevent the leakage of radioactive substances from the canister. The rock provides a natural environment in which the function of the technical barriers is maintained over a very long period. The rock, coupled with the great depth of the deposit, effectively isolates the spent fuel from human beings and the environment.

In Switzerland, the national authority for the storage of radioactive wastes has stated that: [NAGRA, 2009]:

> *Based on current understanding, deep geological disposal is the only method for managing radioactive waste that meets the strict requirements relating to long-term safety. Concepts in which safety relies on continuous monitoring by human institutions are not capable of fulfilling these requirements. The Federal Council and Parliament have therefore decided that all waste arising in Switzerland will undergo deep geological disposal.*

Japanese waste policy, defined in the Specified Radioactive Waste Final Disposal Act 2000, is that HLW should be disposed of in a stable geological formation at a depth of more than 300 m. The vitrified waste (from fuel reprocessing) in canisters is to be encapsulated in metal containers (overpacks) and, once emplaced in a repository, surrounded by a clay buffer material [NEI, 2009].

The United Kingdom Committee on Radioactive Waste Management (CoRWM) was appointed and commissioned in 2003 to review options on the long-term management of the UK's higher activity radioactive wastes. It was asked to recommend the option, or combination of options, that could provide a long-term solution and provide protection for people and the environment.

In 2006, CoRWM submitted its recommendations to Government [CORWM, 2006]. Of 15 recommendations, the first was that the Government should aim to progress geological disposal of higher activity wastes as soon as was practicable. This recommendation was accepted and the Nuclear Decommissioning Authority (NDA) was given the responsibility for implementing geological disposal.

In a 2006 IAEA meeting on the management of spent nuclear fuel [IAEA, 2006a], the following statement was made regarding the difference between direct disposal and disposal following treatment:

> *I would distinguish between the direct disposal of spent fuel, which is one option, the other option being reprocessing and recycling with disposal of the waste.... In fact, whatever scheme is adopted, there will always be a need to dispose of the residual waste in a geological repository.*

However, it is also possible to condition fuel and directly store and dispose of the products. This option may be considered necessary for unstable fuel types.

5.2 Long term interim storage

In March 2004, the Radioactive Waste Management Committee (RWMC) of the OECD Nuclear Energy Agency (NEA) initiated a Working Group on the Roles of Storage (RoSt). The following is taken from the committee's report [NEA, 2006].

Storage has long been incorporated as a step in the management of spent nuclear fuel (this is, of course, the current situation in Norway). Article 2 of the Joint Convention [IAEA, 1997] defines storage as the holding of spent fuel or radioactive waste in a facility that provides for its containment, with the intention of retrieval. According to this definition, and conventionally, storage is an interim step within a waste management strategy in which the ultimate step is still to be done. Typically, stores have been designed and licensed for periods of decades. However, in more recent years, the question has been asked whether storage periods might be extended to provide longer term care of the SNF or waste.

Storage is done to satisfy one or more of several waste management needs, the more important of which include:

- The development and maintenance of a materials backlog; this may be important economically to ensure the continuity of process operations.
- The provision of hold-up capacity to allow radioactive decay, thus reducing radiation and heat generation levels in the waste.
- The accumulation of sufficient quantities of waste to enable economic treatment, transport, recovery etc.
- The provision of hold-up capacity while optimum methods are evaluated for treatment, further storage and disposal.
- The storage of materials that, while not immediately required, have some potential future use or value, and have therefore not been defined as waste.

Extensive experience and technical knowledge related to storage exists in OECD member countries. Storage activities are firmly regulated and the principles for regulation are developed at international level. However, maintaining good quality long-term storage is a complex technical activity that presents ongoing challenges. To maintain satisfactory storage, the facilities must be carefully planned and operated to avoid degradation of facilities, to maintain supervision and to avoid loss of materials and radiation exposures. Storage facilities are also vulnerable to terrorism. Thus, safety and security are not automatic characteristics of storage. Rather, continued safety and security rely on a suite of technical and organisational requirements. To be effective in the long term, these technical and organisational requirements must be underpinned by secure financial resources and an ongoing political and societal commitment to take care of the waste.

In recent years, there have been developments that lead, in some countries, to use storage, or to consider storage, for roles that are beyond those conventionally considered. The most significant development is the discussion of alternative strategies for the long term management of spent nuclear fuel and long-lived radioactive waste, i.e. that geological disposal is not necessarily the end point or that it might be implemented only after an extended period. This may lead to long term storage (i.e. 100 years or more) to provide time to investigate alternatives, to make a decision on end points, or even to wait for developments in science and technology. Some stakeholders consider that long term storage itself is an acceptable endpoint.

Various motives have been put forward for extending storage periods, including:

- Immediate and practical reasons, for example, related to public acceptance.
- Future and strategic reasons, for example, related to possible developments of regional or multi-national solutions or developments in technology.

Different stakeholders may or may not favour longer-term storage. The reasons used to support their position may be partly based on subjective judgment, and their opinions concerning the future and how this generation should limit the liabilities that it will leave for future generations.

If long term storage is being considered, then the technical challenges associated with this strategy must be taken into account. These challenges will increase the longer into the future that storage is contemplated. Besides the technical challenges, ensuring control and maintenance has also to be ensured. The most optimistic assumptions and regulatory requirements about the duration of institutional control periods for waste management facilities do not go beyond 300 years. In any case, these periods are considerably shorter than the period for which long-lived waste present a hazard and needs to be managed.

Consequently, it is important to define the time period for which storage is expected under the strategy. This is needed to enable design, to allow estimation of costs, for licensing, and to indicate the organisational commitment entailed. To maintain the technical assurances of safety and security, it is essential to provide secure financial resources and stability for the organisations and agencies that are charged with carrying out and supervising the storage. This will require continued political and societal commitment, and, also, national economic stability to maintain these organisations and resources. Such factors become harder to guarantee the further into the future that is considered.

If a policy of long term storage is adopted in view of possible future developments, for example solutions based on technology development or multi-national facilities, then the country adopting that policy must work actively to investigate and develop these possible solutions. Waiting, on its own, is not enough.

By definition, storage cannot be an end point for radioactive waste management. Further, it is observed that planning for very long periods of storage, involving multiple renewals of storage facilities is unrealistic and introduces significant uncertainties over which the present generation can have no control. One of these uncertainties is the possibility that such storage might become the endpoint by default, which would be unsatisfactory.

If a responsibility is handed to future generations, then so must the means to deal with the responsibility, so that future generations can at some point discharge the responsibility. This is a matter of financial and intellectual responsibility. It is the duty of the current generation to understand the problem and provide solutions according to current understanding now.

An "open" solution, such as indefinite storage, is not sustainable, because it implies unquantified (and probably unquantifiable) impacts and use of resources. To be sustainable, it is essential to define a complete waste management strategy that does not rely on speculations concerning future societal, scientific or technological developments. (According to the 1987 Brundtland Report, sustainable development is characterised by "Meeting the needs of the current generation without compromising the ability of future generations to meet their needs"). To be complete, the strategy must have a well-defined endpoint and the path (or alternative paths) to reach that endpoint must be specified. To be successful both this and future generations will need to continue working to ensure storage can be ended at an appropriate time.

Future generations may choose to follow the waste management strategy that is defined today, to amend it or to adopt some alternative strategy. In making their decisions they will take into account the increased knowledge that they may then have, and their constraints and preferences. If flexibility is built into a waste management strategy, then future generations will be able to exercise more judgment for themselves. This, however, should not distract the current generation from the importance of its responsibility to implement as much of the plan as can be done now.

5.3 Reprocessing

Reprocessing is the only process that has been used in the closed fuel cycle on an industrial scale. Up to the end of 2004, the total amount of worldwide SNF was close to 268,000 tonnes of heavy metal, of which 90,000 tonnes has been reprocessed [IAEA, 2008a]. The majority of this fuel has been processed using the PUREX process, in which the fuel rods are cut into suitable lengths and then dissolved in nitric acid. Following dissolution, uranium and plutonium are extracted into to an organic phase by intensive mixing with an organic solvent - 30 percent tributyl phosphate (TBP) in kerosene is commonly used - while the fission products remain in the aqueous (nitric acid) phase. Further process steps enable the subsequent separation of uranium and plutonium from one another. The fission products are mixed with glass forming elements to form a vitrified waste product, which is considered suitable for geological disposal. A conceptual diagram of separation by solvent extraction is shown in Figure 5-2, while Figure 5-3 presents a flow diagram of the PUREX process.

Reprocessing of civil SNF is currently performed in France, India, Japan, the Russian Federation and the UK. After having developed the closed fuel cycle in the early days of nuclear power, the USA switched to a once-through cycle in 1978 mainly because of proliferation concerns (i.e. the production of separated plutonium). Early in 2006, a major political transition occurred with the launching of the Global Nuclear Energy Partnership (GNEP) initiative. A return to the closed fuel cycle was proposed for both domestic reasons (especially regarding the optimization of the geological repository for ultimate waste) and for the implementation of a multinational approach to the fuel cycle. However, given the change of government in 2009, the future of this initiative is not known.

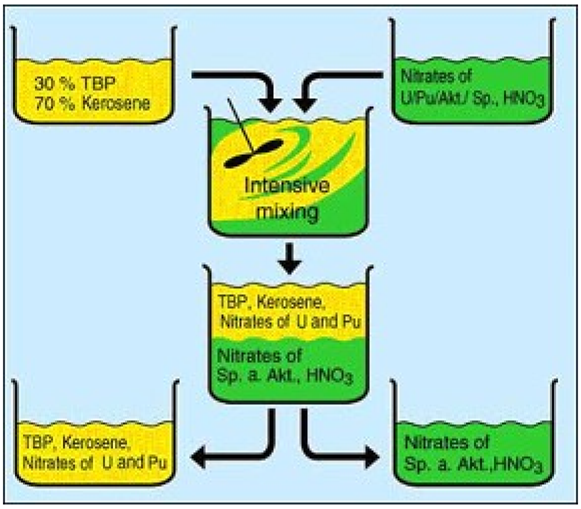


Figure 5-2. Conceptual diagram of separation by solvent extraction

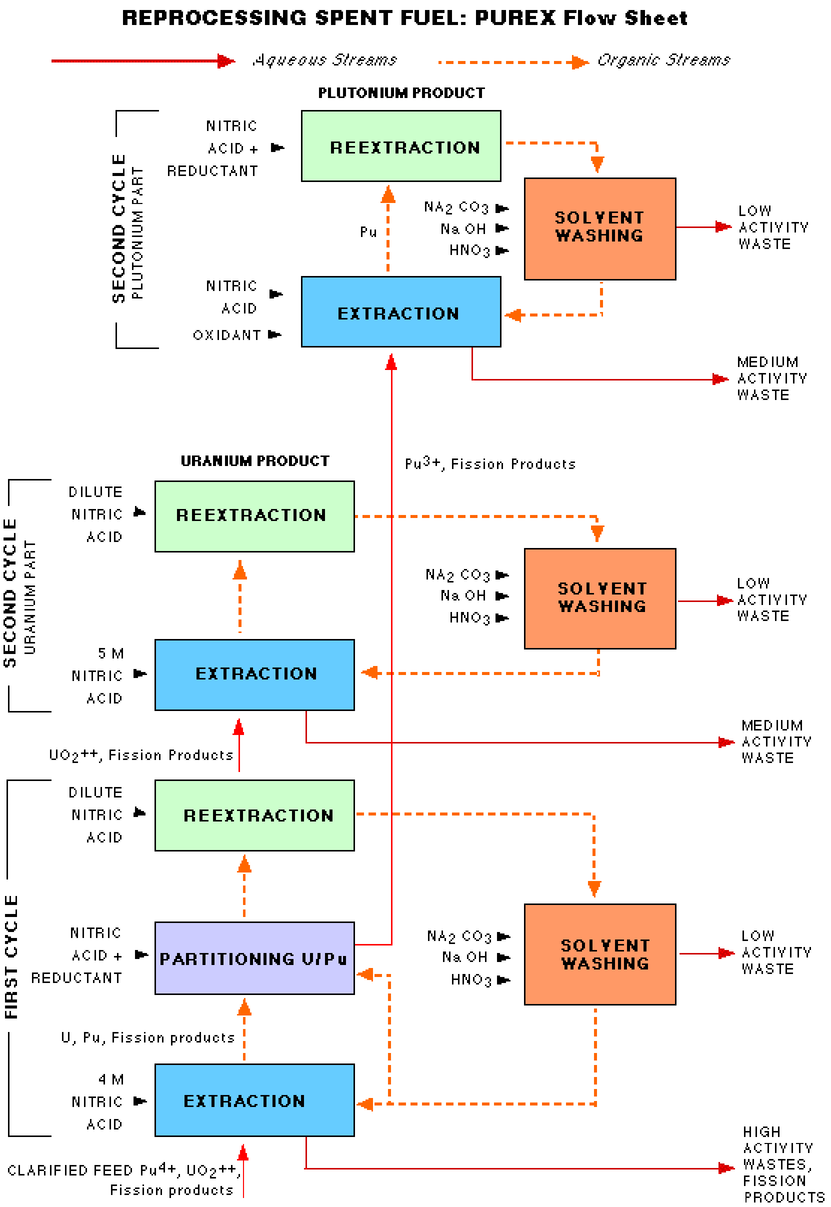


Figure 5-3. PUREX flow diagram

Proliferation risks are only one of the reasons why many countries have decided against the reprocessing option. Other reasons include the risk of discharges to the environment from reprocessing facilities and the costs of building, operating and decommissioning the required facilities.

Both metallic uranium fuel and fuel clad in aluminium can be reprocessed using the PUREX process. As discussed in Chapter 6, relevant examples include treatment of research reactor fuel at the Savannah River Site in the USA, MAGNOX and Swedish R1 reactor fuel at Sellafield in the UK, and Australian research reactor fuel in France.

Currently, research into advanced reprocessing technologies is being performed in many countries with large nuclear power programs. The wide range of research reflects the equally wide range of objectives. These range from modifications and developments of the PUREX process to new technologies which are still at the conceptual stage. To generalise these developments, it may be stated that next-generation reprocessing plants are likely to be based on aqueous extraction processes that can be designed to a country specific set of spent fuel partitioning criteria for recycling of fissile materials to advanced light water or fast reactors.

The next generation (Generation 3) technologies are based on aqueous separation methods aimed at co-management of U and Pu, or U-Pu-Np. These changes would avoid the separation of a pure plutonium stream to improve proliferation resistance. Some of these developments are at an advanced level of technological maturity, and could be deployed in the next generation of industrial scale reprocessing plants, which would be built to replace existing facilities. The earliest date mentioned for industrial scale implementation of these technologies is 2020 – 2025.

At a more experimental stage are aqueous processes using new extractant molecules which would allow selective separation of minor actinides for interim storage, pending a decision regarding their transmutation in fast reactors or accelerator driven systems.

Electrometallurgical (or pyrochemical/pyrometallurgical) methods allow for the reprocessing of different types of highly radioactive fuels, such as metals, carbides, oxides or nitrates, with high contents of fissile materials. Such methods are foreseen, for example, in Generation IV reactors such as the Very High Temperature Reactors, Gas-cooled Fast Reactors or advanced Sodium Cooled Fast Reactors. These processes are the main alternative to aqueous methods, and are the subject of a renewed development effort on an international scale. The generic principle of these processes consists of dissolving the fuel in a bath of molten salts (chlorides, fluorides etc) at temperatures of several hundred degrees Celcius, then separating the desired species using techniques such as liquid metal extraction, electrolysis or selective precipitation (Figure 5-4). These processes are of interest because of their abilities to dissolve refractory fuel materials, their suitabilities for reprocessing fuel immediately after unloading, the potential compact nature of the facilities, and the assumed suitability for group management of the actinides. Significant development work has been done in the USA (Argonne National Laboratory) and the Russian Federation (RIAR Institute) on metal and oxide fuels respectively, up to and including the construction of pilot plants. Major uncertainties remain however about the implementation of pyrochemical processes, including the actinide recovery factor and their implementation at industrial scale.

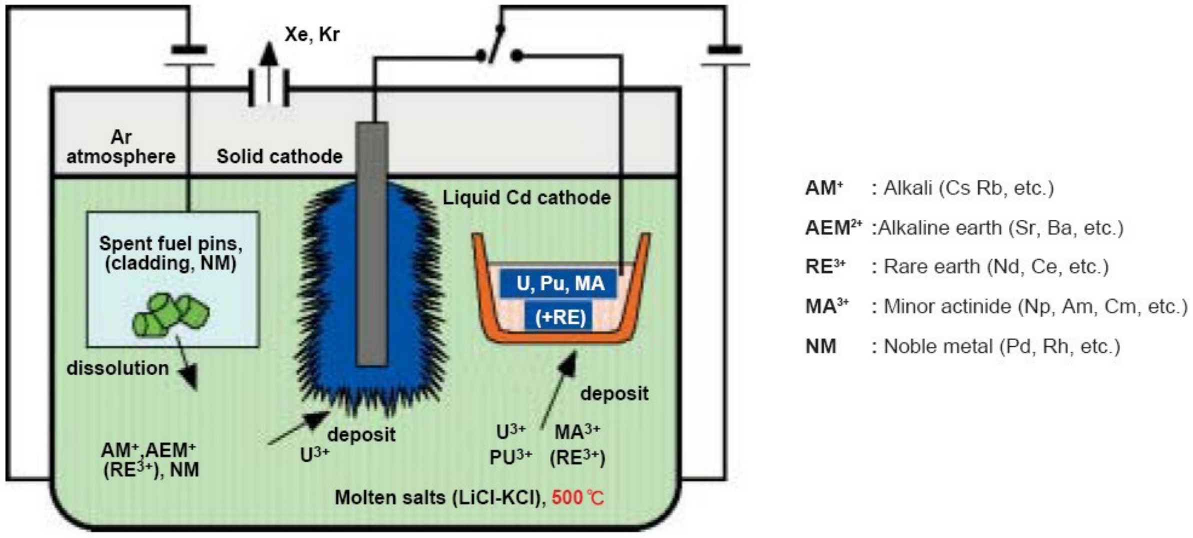


Figure 5-4. Diagram of electrorefining

Electrorefining is currently being used in the USA for the conditioning of metallic spent fuel from the EBR-II reactor. This process is discussed in more detail in Chapter 6.2.

Notwithstanding the developments in advanced reprocessing techniques, the IAEA report on advanced reprocessing [IAEA, 2008a] stated that:

> *Conventional reprocessing technology is expected to continue to play an important role in the medium term. ...there are still a number of issues to be addressed in preparation of the forthcoming new generations (Generations III and IV) of facilities aimed at replacing the operating plants at the end of their life. ...it will be several decades before Generation IV plants are commercially available. ...Apart from the technical development required, all the other issues relating to the industrialisation of emerging technologies need to be addressed, such as licensing, socio-political issues and the decision as to whether to commercially invest in them.*

5.4 Other options

One option for spent fuel which meets certain criteria is return to the country of origin. The U. S. Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance- and the Russian Research Reactor Fuel Return (RRRFR) programs allow return of high enriched fuel that was enriched in the USA or the Russian Federation (ex USSR). Another option for spent fuel management that has been used previously is that two countries exchange fuel if such an exchange is beneficial to both. These options are discussed in more detail in Chapter 6.

6 INTERNATIONAL EXPERIENCE AND RECOMMENDATIONS FOR TREATMENT OF METALLIC URANIUM FUELS AND ALUMINIUM-CLAD FUELS

6.1 R1 Reactor fuel, Sweden

The R1 reactor operated from 1954 until 1970, at KTH in Stockholm. The fuel, 4.8 tonnes, was natural metallic uranium in aluminium cladding. The fuel was thus of the same type as the Norwegian JEEP I and early HBWR fuel. The main part of the fuel originated in Sweden, while a minor part was Canadian.

After the reactor was closed down, the fuel was transported to Studsvik for storage. The major part was stored in a dry storage facility (AT) with the remainder in a cooling pond in the fuel storage building (FA). In the FA, the fuel was put into aluminium containers. Ownership of the fuel was transferred to AB SVAFO according to the “Studsvik Act”.

In 1987 it was noticed that gas was leaking from one fuel container in the storage pond in the FA. A subsequent investigation in 1988 revealed that water had been leaking into the container, and black corrosion products (together with water) were observed [Holmér et al, 1988]. A special outer container was designed and constructed for storing the leaking fuel. Plans were made for reducing possible hydride production [Björner et al, 1990; Jonsson and Holmér, 1993].

In parallel, a process started to find a solution for future safe storage and final disposal for all the R1 fuel, but with priority given to the non-corroded elements. As a result, a contract was signed in 1998 between AB SVAFO and British Nuclear Fuel limited (BNFL), Sellafield, UK, to transport and treat (reprocess) the R1 fuel. The outline of the contact was:

- That BNFL should reprocess the fuel in the Sellafield MAGNOX reprocessing plant.
- That the retrieved plutonium would be transferred to Oskarshamn KraftGrupp AB(OKG) for MOX fuel fabrication, to be used in OKG reactors and then transferred into the Swedish route for used fuel.
- That the vitrified HLW would be retained in Sellafield, as it did not comply with the Swedish system for handling HLW. The depleted uranium and plutonium-containing-material (PCM) waste was to be sent back to Sweden. The amount of the PCM waste was to be increased (based on radiotoxic index calculations) to compensate for the non return of the vitrified HLW.

In order to meet BNFL’s receival requirements, The Hot Cell Laboratory planned and accomplished a test campaign to verify the methods to pre-treat the fuel. This comprised cutting off the top and bottom aluminium parts of the rods and cutting them into suitable lengths for the transport vessel [Holmér and Sundström, 1996].

Later in 1998, an application was sent to the Competent Authority, SKI, by AB SVAFO. In 1999, SKI made a review statement to AB SVAFO [SKI, 1999 a] and the Swedish Government [SKI 1999 b] [Holmér and Sundström, 1996] after obtaining SSIs opinion [SSI, 1999]. In its submission to the Swedish government, SKI made the statement that, in its opinion, from safety considerations, metallic uranium fuel is unsuitable (*olämpligt*) for direct disposal in a geological repository of the type under consideration in Sweden. Specific reasons were that metallic uranium reacts with water and air to form UO_2 and UH_3; the latter and metallic uranium are pyrophoric. SKI concluded that:

> *Corroding metallic uranium fuel is a serious safety problem.*

However, the Swedish Government of the time declined to make a final decision regarding this fuel. Following public criticism of Sellafield by the Minister of the Environment, AB SVAFO rescinded its application in order to investigate other treatment and disposal options [SKB, 2005]. These options included:

1. Fuel exchange

Fuel exchange is an option when two countries would both benefit from the exchange. In the case of the R1 fuel, such an exchange with German fuel was considered the best of the considered options based on experience of a successful exchange in 1985. This earlier exchange involved 57 tonnes of Swedish fuel from power reactors, which would be reprocessed in the French La Hague facility, and 24 tonnes of MOX fuel from Germany.

2. Intermediate storage in a modern facility

This option was not recommended as it does not constitute a final disposal solution.

3. Direct disposal in a repository

Final disposal of metallic uranium fuel was not recommended because of the instability of the fuel with respect to reaction with water.

4. Reprocessing in the USA, Russia and the UK (Gudowski report)

A report on the possible options for treatment of the R1 fuel in the USA and Russia was prepared by Professor Gudowski of KTH. Treatment of civil spent fuel is not allowed in the USA. In Russia, four facilities were investigated. Although treatment of the fuel was technically feasible, in its review of the report, SKB concluded that it would be politically impossible to ship Swedish fuel to Russia.

In conclusion, the report stated that the most realistic option for the R1 fuel was reprocessing by BNFL in the UK.

In the meantime, in 2002 all non-corroded R1 fuel stored in the pond in the FA was transferred to the dry storage AT for re-packing into new containers. Only the special container with the corroded fuel was left in the FA.

In 2004, after consultations with SKI and SSI, AB SVAFO decided to make a new application. SSI and SKI were now entitled by the government to take the formal decisions on authority level, which made the process less complicated from a political point of view. The then Minister of the Environment was in favour of the reprocessing solution, but had no formal role in the decision making.

In 2005, new applications were sent to SKI and SSI for permission to send the fuel to Sellafield for reprocessing. In the autumn, AB SVAFO received formal permission, by a “Government Letter”, to transfer (export) the fuel to Sellafield and take back (import) the waste formed during the process [SSI, 2005].

Planning for the pre-processing and transport to Sellafield was made by Studsvik Nuclear AB on behalf of AB SVAFO [Eriksson, 2005]. The accomplishment of the project is described in [Eriksson, 2007]. Costing and prices are not shown in this report due to commercial reasons. The fuel was cut and re-packaged before transport (Figure 6-1).

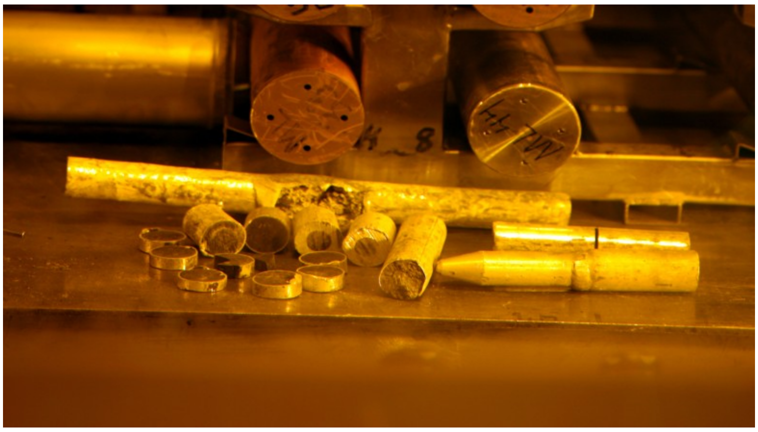


Figure 6-1. R1 fuel before shipment to Sellafield

In 2006, AB SVAFO started a project to manage the corroded fuel [Nordlinder and Ekberg, 2009]. The purposes of this project were to dry out the material, to uncover and expose uranium hydride, and subsequently to separate metallic uranium from corrosion products. The work required a hot cell and specially-designed equipment. Figure 6-2 shows the appearance of corroded R1 fuel after opening of a storage canister.

All R1 fuel that was sent to Sellafield is now processed and the waste fractions have been sent back to AB SVAFO, for storage until the final repository is commissioned.

Figure 6-2. Corroded R1 fuel

6.2 United States

Tens of thousands of tons of radioactive material were reprocessed at the Savannah River Site (SRS), mainly for defence purposes, from the mid-1950s to the beginning of the 1990s.

In October 1976, a Presidential directive was issued to indefinitely suspend the commercial reprocessing and recycling of plutonium in the U.S. Later, in 1992, The US stopped reprocessing of spent nuclear fuel for production purposes. Since then, limited processing of fuel that presented environmental, safety or health risks has been performed. This directive has resulted in many extensive research programs to study alternatives to reprocessing for management of US spent fuel.

A 1997 overview of the United States Spent Nuclear Fuel Program summarised the then position with respect to fuels relevant to the current study [Hurt, 1997]. DOE stored over 150 types of spent fuels. The current strategy was to place all SNF in dry storage and in “road ready” condition for eventual transport to, and ultimate disposal in, a geological repository. Some of these fuels may not be suitable for direct disposal because they are chemically reactive or prone to corrosion. Reducing the volume of materials to be disposed of would greatly reduce the requirements for disposition casks.

For non-sodium bonded fuels, chemical separation at the Savannah River Site was specified as a contingency to be used if no acceptable new treatment/packing technology was ready to be implemented by the year 2000.

On the subject of chemical reactivity, it was noted that the current regulations required that packages destined for the national repository shall not contain explosive or pyrophoric materials or chemically reactive materials in an amount that could compromise the ability of the underground facility to contribute to waste isolation or the ability of the geologic repository to satisfy the performance objectives.

In the mid to late 1990s, work was done in the US to investigate non-reprocessing options for the treatment of aluminium spent nuclear fuel. The following is based on the report of this work [Levenson and Crowley, 1998].

The Savannah River Site is responsible for managing DOE's inventory of aluminium SNF from foreign and domestic research reactors. These responsibilities include receiving and storing the fuel, processing it as necessary to put it into a "road ready" condition for eventual transport to a repository and providing for interim storage of the "road ready" product until a repository is ready to accept it.

In 1995, a Research reactor Spent Nuclear Fuel Task Team was established by the DOE to help develop a technical strategy for disposal of the Al SNF. The team was to evaluate alternative treatment and packaging technologies that could be used in place of conventional reprocessing to treat for disposal the fuel in a safe and cost-effective manner. This need for alternative methods arose from the decision by the Energy Secretary in 1992 to phase out reprocessing at the DOE's sites.

The treatment technologies evaluated by the team fell into one of three categories:

1. Direct disposal technologies, which involve no processing except for drying (*most of the SRS fuel was stored in water*).
2. HEU dilution technologies, in which fuel is physically or metallurgically diluted with depleted uranium. *(This option is not relevant for the Norwegian Al-clad fuel as the maximum enrichment is 3.5 per cent.)*
3. Advanced treatment technologies, in which the fuel is processed to produce more acceptable waste forms for repository disposal than is possible for either of categories 1 and 2, and to reduce the volume of waste for disposal.

The task team used a combination of qualitative and quantitative methodologies to screen and rank the alternative treatment options, and performed a sensitivity analysis to eliminate options that were less likely to be implemented successfully because of technical, cost or scheduling difficulties.

As the primary treatment option, the team recommended direct co-disposal treatment. This involves the placement of dried Al-fuel into a canister in a larger package containing high-level waste glass logs. As a parallel option, melt-and-dilute was recommended, in which the fuel is melted and diluted with depleted uranium. As a back-up option, the team recommended electrometallurgical treatment (electrochemical reprocessing) because this technology is fundamentally different from the others and thus offered some protection against unforeseen technical or licensing problems.

As part of its efforts to prepare an environmental impact statement (EIS) for disposing of the fuel, the Savannah River Office of DOE asked the National Research Council (NRC) to review the alternative treatment options. The statement of task for the study included, among others, examination of the set of technologies chosen, and identification of other alternatives that DOE might consider. Relevant findings of the NRC's review are summarised below.

The NRC believed that the team had succeeded in identifying a reasonably complete set of treatment alternatives, but noted that one option – chloride volatility treatment – may have been incorrectly eliminated, and that options had not been considered for declad, uranium metal fuel. The task team had identified this fuel as a candidate for reprocessing because:

> *metallic uranium may not be suitable for disposal in a repository. However, if reprocessing is not possible for policy reasons, there is no obvious treatment and disposition pathway for this fuel.*

In response to the question on whether the treatment options were likely to work as described and produce acceptable waste forms, the NRC review reported that melt-and-dilute is more technically demanding that direct co-disposal treatment and would require a more significant infrastructure, including hot cell space, a melter, and an off-gas treatment system. The fuel must be melted at temperatures up to about 1000 °C, which will release volatile fission products that must be recovered and disposed of. Advantages of the technique included good control over waste form composition and waste volume reduction. It was also reported that there was not enough information about any of the advanced treatment technologies to select a back-up option, in particular about electrometallurgical treatment.

The NRC also considered that DOE should have given more careful consideration to the conventional reprocessing option. There appeared to be several technical advantages to this option over the others considered by the task team. The technique has been demonstrated to work for Al spent fuel, the cost and risks were well known, the necessary facilities were currently in operation at SRS, and the waste form (borosilicate glass) will likely be acceptable for disposal at the repository.

In its concluding observations, the NRC remarked that a spent fuel disposal program is a systems problem in the classic sense: it involves several interacting components, each associated with different programmatic factors (e.g. cost, time, safety, policy constraints), multiple responsible parties, and different levels of uncertainty. A phased strategy was required, with the primary objectives to maximise the probability of program success, minimise overall costs, and protect the program against down-side risks from changes over which it has little or no control.

Further, the NRC stated that:

> *There does not appear to be a technical basis for rejecting conventional reprocessing as an option for aluminium spent fuel. ... (It) is a proven and reliable spent fuel treatment technology based on over 300 plant-years of operation worldwide.*

A cost study suggested that conventional reprocessing was cost-effective compared with direct co-disposal and melt-and-dilute:

> *Although it is difficult to make quantitative comparisons between a proven and an unproven technology, it is clear that the cost, performance and safety of unproven technologies have much greater uncertainties than those of a demonstrated technology such as reprocessing.*

> *The concern with conventional reprocessing appears to be mainly one of policy and is related to the use of reprocessing for waste management rather than any specific concern about reprocessing this particular fuel type.*

The above technical developments were accompanied by political decisions in connection with the U. S. Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Program [Lyman, 2004]. (This program is discussed in more detail in Chapter 6.6.) A 1996 DOE Record of Decision (ROD) committed to "avoid indefinite storage of this SNF in a form that is unsuitable for disposal". To this end, DOE further committed to "aggressively pursue one or more new (treatment and packaging) technologies that would put foreign research reactor spent fuel in a form or container that is eligible for direct disposal in a geologic repository".

Subsequently, in its 2000 ROD of Savannah River Site spent fuel management, DOE decided to implement the melt-and-dilute method to manage most of the Al-clad fuel at SRS. However, DOE abruptly cancelled the program in 2002 without proposing an alternative.

Following the various technical developments and changes in policy since the 1992 decision to halt reprocessing outlined above, the wheel has now turned full circle, as reported by SRS [SRS, 2009]:

> *DOE plans to use conventional processing, through the H canyon chemical separations facility, as the final disposition of all aluminium-clad SNF currently stored at SRS as well as planned receipts of foreign and domestic research reactor SNF through 2019. ... Conventional processing of the SNF will provide additional uranium for the HEU Blend Down Program for use in commercial production of electricity and produce liquid waste to be vitrified in SRS's Defense Waste Processing Facility.*

An electrorefining (or pyrochemical) process is being used for the conditioning of metallic spent fuel from the EBR-II reactor [National Academies, 2000; USDOE, 2006; IAEA, 2008a]. The fuel consisted of pure metallic uranium or metallic uranium alloyed with either zirconium metal, a mixture of noble metals, or plutonium-zirconium metal, encased in stainless steel cladding and bonded to the cladding with sodium. The purpose of the sodium was to improve heat transfer. Since sodium reacts with water, aqueous reprocessing presents serious hazards, and thus the pyrochemical method was developed.

The process employs a high temperature electrolytic cell containing a molten LiCl-KCl salt and steel electrodes (Figure 6-3). One electrode contains chopped spent fuel, which is electrochemically dissolved in the molten salt when a voltage is applied to the system. Oxidation of metals from the fuel to chlorides in the salt occurs at the anode, resulting in the formation of sodium chloride, uranium chloride and various fission product and transuranic chlorides. Simultaneously, uranium is deposited on a solid metal cathode immersed in the molten salt. This recovered uranium is stored for use as new fuel.

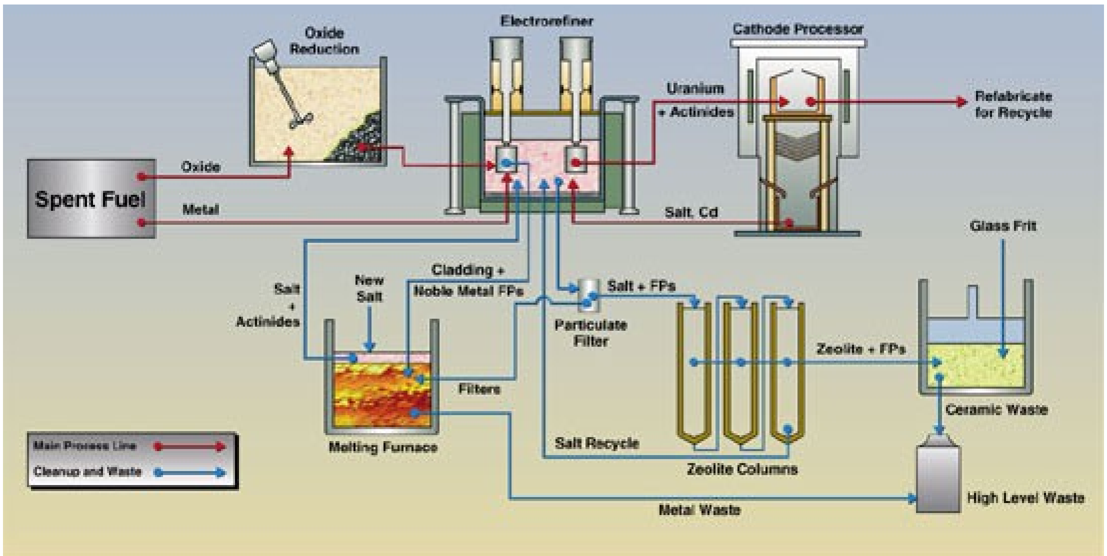


Figure 6-3. Pyroprocessing flow diagram

The transuranics and the active metal fission products are left in the salt for eventual incorporation in a ceramic waste form. Noble metal fission products are melted together with the stainless steel cladding hulls to produce a metallic waste form.

Concerning the applicability of this process for other DOE-owned fuels – in particular Al-clad fuels from Hanford, discussed below - the National Academies report stated that:

> *It is unlikely that (Al-containing) fuel could be processed directly by the electrometallurgical process without additional head-end treatment to remove the aluminium. Aluminium's tendency to form low-melting eutectic mixtures and volatile species would present a significant challenge for containment and effluent treatment in the electrorefining process.*

In the 1990s, over 2000 metric tons of US Government-owned spent fuel was stored in the KE and KW wet storage basins at DOE's Hanford site. Most of this fuel was from the N reactor, which operated from 1963 until 1987, producing materials for the US national defence program. The majority of the fuel comprised metallic uranium clad in Zircaloy-2; in addition there was a relatively small amount of single pass fuel, which consisted of metallic uranium sealed in, and metallurgically bonded to, aluminium alloy cladding. Although the majority of the fuel was clad in Zircaloy, which is stable against water corrosion, the subsequent treatment of this fuel is described in some detail as the Al-clad fuel was included with the predominant Zircaloy-clad fuel.

By this time there were environmental and safety concerns associated the KE basin arising from, among others, the presence of broken and corroding fuel and build-up of radioactive sludge on the bottom of the basin. In 1996, the US Environmental Protection Agency (EPA) issued a Record of Decision (ROD) on the management of these fuels [USEPA, 1996]. The chosen method was removal of the spent fuel from the basins, vacuum drying, conditioning and sealing the SNF in inert-gas filled canisters for dry vault storage in a new facility for up to 40 years, pending decisions on ultimate disposition.

After loading the fuel into multicanister overpack (MCO) containers, and then draining the MCOs, the fuel is dried under vacuum at 50 °C, and the canisters are flooded with inert gas and sealed. The canisters are transported to a dedicated storage building, and are then subjected to further conditioning: heating the canisters to 300 °C to remove water that is chemically bound to the SNF and canister corrosion products, and to dissociate, to the extent practicable, any reactive uranium hydride.

Principal advantages of the drying/passivation (conditioning) with dry vault storage alternative are that it would accelerate removal of SNF from aging facilities in proximity to the Columbia River, would result in passive vault storage of dry SNF requiring only minimal surveillance, would retard continued degradation of the SNF and would reduce or eliminate reactive uranium hydrides in the SNF. Principal disadvantages of this alternative are that the construction of new facilities would be required, and some uncertainty exists in the chemical state of the SNF and sludge and, therefore, in the extent to which drying and passivation processes would be required. However, defense-in-depth measures will be engineered to assure safety of the process. Moreover, characterization of K Basins SNF is presently being conducted to address these uncertainties which may result in a more cost-effective conditioning process.

Six other alternatives - which were rejected - for the management of SNF from the K Basins at the Hanford Site were analysed. These are repeated here to illustrate the issues that must be considered in any spent fuel management program.

1. No action alternative: Under this alternative DOE would continue SNF storage in the KE and KW Basins for up to 40 years with no modifications except for maintenance, monitoring, and ongoing safety upgrades. Consideration of the no action alternative is required by CEQ regulation [40 CFR 1502.14(d)]. The principal advantage of the no action alternative is that it would require no movement of SNF and no construction of new facilities. Principal disadvantages of this alternative are that the K Basins were not designed for an 80-year life (40 years to date and up to an additional 40 years) and would require increasing maintenance of aging facilities with associated potential for increased radiological impacts on workers, would not place the SNF in a safer storage configuration, would not preclude leakage of radionuclides to the soil beneath the basins and near the Columbia River, and would fail to alleviate concerns expressed by regulatory agencies, advisory bodies and the public relative to environmental impacts induced by seismic events.

2. Enhanced K Basins storage alternative: Under this alternative DOE would perform facility life extension upgrades for KW basin, containerize KE Basin SNF and sludge, and consolidate with KW Basin SNF for up to 40-year storage. Principal advantages of the enhanced K Basins storage alternative are that it would remove degrading SNF from the KE Basin, permit deactivation of the KE Basin, and would require no construction of new facilities. Principal disadvantages of this alternative are that the KW Basin was not designed for an 80-year life and would require increasing maintenance of the aging facility. Despite completion of practical upgrades, this alternative would not arrest continued fuel degradation, might result in conditions favorable to the production of reactive uranium hydrides in the repackaged KE Basin SNF transferred to the

KW Basin, and would fail to alleviate concerns expressed by regulatory agencies, advisory bodies and the public relative to environmental impacts potentially induced by seismic events.

3. New wet storage alternative: Under this alternative DOE would remove SNF from the K Basins and provide for up to 40 years of new wet storage in a new facility located on the 200 Areas plateau that meets current design criteria. Principal advantages of the new wet storage alternative are that it would accelerate removal of SNF from aging failities in the proximity to the Columbia River, would make use of a proven storage technology (at least for commercial fuel) coupled with design to modern seismic criteria, and would maintain flexibility for preparing SNF for ultimate disposition. Principal disadvantages of this alternative are that it would require construction expense and continued maintenance, would not prevent the continuation of SNF degradation, and would not eliminate the potential for further hydriding of the SNF.

4. Calcination with dry storage: Under this alternative DOE would remove SNF from the K Basins, calcine it, and provide for up to 40-year dry storage of SNF-oxides in a new cask or vault facility. The principal advantages of the calcination with dry storage alternative are that it would remove the SNF from aging facilities near the Columbia River and that it would convert the SNF into stable oxides, which are readily storable in a dry form and may be suitable without further processing for ultimate disposal in a geologic repository. The principal disadvantage of this alternative is the need to construct and operate a relatively expensive calcining facility.

5. Onsite processing: Under this alternative the DOE would remove and chemically process K Basins SNF and provide for up to 40-year dry storage of the recovered uranium (as uranium trioxide) and plutonium (as plutonium dioxide), and manage fission product waste in tanks with other wastes under Hanford's Tank Waste Remediation System program. Principal advantages of the onsite processing alternative are that it would remove the SNF from aging facilities near the Columbia River, convert uranium (the major constituent of SNF) into uranium trioxide that is readily storable in dry form and for which future use (constituent of power reactor fuel) might be found, convert plutonium to a stable oxide for which a future use (constituent of power reactor fuel) might be found or for which storage in a geologic repository may be suitable without further processing, and convert fission products into a form suitable for storage in a geologic repository. Principal disadvantages of this alternative are the need to construct and operate a relatively expensive separations facility, the plutonium dioxide product would no longer be self-protecting and would require special storage and accountability that in turn may require construction of additional storage capacity, and no immediate need exists for either the separated uranium or plutonium.

6. Foreign processing: Under this alternative, the DOE would remove K Basins SNF, ship overseas for processing, provide for up to 40-year dry storage of returned uranium (as uranium trioxide) and plutonium (as plutonium dioxide), and store vitrified fission product waste, pending ultimate disposition. With the exception that foreign processing would obviate the need for construction of additional processing facilities at Hanford, the principal advantages of the foreign processing alternative are essentially the same as those for onsite processing. Principal disadvantages of the foreign processing alternative are the need to transport the K Basins SNF to a U.S. shipping/receiving port, transload the SNF to ocean vessels, ship the SNF to a foreign port,

transport the SNF to an operating reprocessing plant, and ship the uranium and plutonium products and vitrified high-level waste back to Hanford or elsewhere, as appropriate. Additional disadvantages include issues associated with the U.S. nuclear nonproliferation policy, unfavorable agency and public opinion regarding shipping the degraded fuel off the Hanford Site, costs of new shipping casks, and construction of a new head-end facility at the processing plant. The need for special storage for plutonium product would be the same as in the onsite processing alternative.

6.3 France

Information on French policy for treatment of SNF has been obtained from the Second Review Meeting on the Joint Convention on the safety of spent fuel management [IAEA, 2006b]. The French answers to questions and comments received from other Contracting Parties on its second report for the JC contained the following statements:

> *The EDF* (Electricity de France) *industrial strategy is based on reprocessing of all the spent fuel unloaded.*
>
> *CEA's* (The French Atomic Energy Commission: Commissariat à l'énergie atomique) *reference strategy is to reprocess all research reactor spent fuel.*

6.4 Gas-cooled power reactor fuel: France and the UK

Gas-cooled reactors (GCR) were developed in France and the UK during the early years of civil nuclear power production. The French UNGG (Uranium Naturel Graphite Gaz) was developed independently of and in parallel to the British Magnox design.

The reactors were graphite moderated, cooled by carbon dioxide, and fueled with natural uranium metal. The fuel cladding materials were magnesium-zirconium alloy in the UNGG and magnesium-aluminium in Magnox. Both claddings react with water, making short-term reprocessing of the fuel essential.

Thus, there were no long term storage plans for GCR fuels, as reprocessing was an integral part of the fuel cycle. In the UK, not all Magnox fuels have yet been reprocessed, although the current reference strategy is to reprocess it all, in the Sellafield Magnox reprocessing plant [CORWM, 2009]. However, since the Sellafield plant is old, there is no guarantee that it will be running long enough to process all the fuel. Hence, the owner of the fuel, the UK Nuclear Decommissioning Authority (NDA), is reviewing alternative management strategies for use should the reference strategy be unavailable. Three strategies are being considered:

1. ***Encapsulation of the fuel in a suitable matrix (e.g. a polymer or cement), followed by its geological disposal.***

   It is noted here that a substantial R&D program would be required to find a suitable wasteform for disposal.

2. Reprocessing through THORP (The Thermal Oxide Reprocessing Plant), with geological disposal of the HLW

This option would require a new dissolver vessel for THORP, which would be costly and would disrupt the THORP program.

3. Drying the fuel, placing it in canisters for dry storage, then preparing the fuel for disposal, and geological disposal

This option is also not straightforward. Drying of metal fuel has been shown to be viable at Hanford in the USA (as discussed above, in Chapter 6.2), but considerable R&D would be needed to show that the dried fuel would be in a suitable form for disposal.

In the event that the Magnox plant becomes unavailable and none of the above three methods have been shown to be viable, the fallback position would be continued storage while R&D proceeds.

6.5 UK Dounreay Fast Reactor (DFR) breeder fuel

A fast breeder reactor (FBR) breeds fuel (from fertile material) by producing more fissile material than it consumes. After the initial fuel charge such a reactor is refuelled from reprocessed fuel.

The Dounreay Fast Reactor (DFR) operated at the United Kingdom Atomic Energy Authority's (UKAEA) Dounreay site from 1959 until 1977. The reactor core was fuelled initially with uranium metal fuel stabilized with molybdenum and clad in niobium. The core was later used to test oxide fuels for the Prototype Fast Reactor (PFR, operation from 1975 to 1994) and provide experimental space to support overseas fast reactor fuel and materials development programmes.

Fuel from the Dounreay fast reactors was reprocessed on site at a dedicated plant, named D 1206, where 30 tonnes of irradiated fuel was reprocessed between 1980 and 1996. In September 2000 the plant was shut down after a leak was discovered in a dissolver. The cost of replacing the equipment and re-opening the plant was considered to be prohibitive.

The majority of the breeder fuel elements from DFR had not been processed at the time the reprocessing plant was shutdown, and has been since stored on site. This fuel consists of metallic uranium clad in stainless steel.

The preferred strategy for the DFR breeder fuel is to process it in the Sellafield Magnox reprocessing plant [CORWM, 2009]. The UK Nuclear Decommissioning Authority (NDA) has asked Dounreay and Sellafield to submit proposed changes to their Lifetime Plans to implement this strategy. Further work is required to confirm that the preferred strategy is fully viable before is can be designated as the reference strategy. One or more contingent strategies will also be developed, but these are not specified in the report.

6.6 The U. S. Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Program

In 1986, to further encourage foreign research reactors to covert to Low Enriched Uranium (LEU) fuel, the US Department of energy (DOE) "Off-Site Fuels Policy" was extended to include the acceptance of foreign spent nuclear fuel containing uranium enriched in the United States. The US accepted foreign research reactor spent fuel until the program expired (in 1988 for HEU fuels and 1992 for LEU fuels) [IAEA, 2008b].

Following expiration of the program, requests were made from several sources to the US DOE to accept US-origin spent fuel from foreign research reactors. Towards the same end, the Director General of the IAEA, Hans Blix, wrote letters to the Secretary of the US DOE (in July 1993) and to the Minister of Atomic Energy of the Russian Federation (February 1995), suggesting that these major partners in the Reduced Enrichment Research and Test Reactor (RERTR) program could facilitate the non-proliferation goal of RERTR by taking back foreign research reactor fuel.

A Record of Decision was published by DOE in May 1996 to re-start the U. S. Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Program, with a deadline of May 2006 for eligible fuel to be discharged from reactors and a deadline of May 2009 for fuel to be received in the US. In 2004, a revised Record of Decision extended these deadlines to May 2016 and May 2019, respectively. This extension allows time for further development of LEU fuels and planning back-end solutions to the fuel cycle [IAEA, 2008b].

The goal of the program is to reduce the threat of nuclear weapons proliferation while enjoying the benefits of nuclear technology. The program recovers nuclear materials which could otherwise be used in weapons, and reduces (eventually eliminates) HEU from worldwide commerce. It allows time for countries with spent fuel to resolve their own disposition [Messick, 2006].

Eligibility for returning SNF in the FRRSNF program is dependent (among others) on the following conditions [IAEA, 2008b]:

- The reactor should use fuel that was enriched in the United States.
- The reactor start-up was before May 1996.
- The country must agree to participation based on the country's economic status as determined by the World Bank.
- The country must enter into a contract outlining detailed responsibilities.
- Reactor status:

The reactor must be operating on or converting to LEU fuel.

Shutdown.

The reactor can be operating with HEU, but must formally agree to convert to LEU within the policy period.

The reactor must be operating with an HEU core for which no suitable LEU fuel exists.

As of May 2009, 47 fuel shipments had taken place. Twenty seven countries have participated, returning a total of over 7000 fuel elements and a total of over 1215 kg HEU fuel. Twenty eight of the shipments contained over 5900 aluminium-based fuel elements, which were sent to the Savannah River Site (SRS). The countries that have shipped Al-based fuel are Argentina, Australia, Austria, Brazil, Canada, Chile, Columbia, Denmark, Germany, Greece, Indonesia, Italy, Japan, Netherlands, Phillipines, Portugal, Spain, Sweden, Switzerland, Taiwan, Thailand, Uruguay, and Venezuela [IAEA, 2008b].

SRS has reported the following on the final disposition of the Al-based fuel stored on site [SRS, 2009],

> *DOE plans to use conventional processing, through the H canyon chemical separations facility, as the final disposition of all aluminium-clad SNF currently stored at SRS as well as planned receipts of foreign and domestic research reactor SNF through 2019. ... Conventional processing of the SNF will provide additional uranium for the HEU Blend Down Program for use in commercial production of electricity and produce liquid waste to be vitrified in SRS's Defense Waste Processing Facility.*

6.7 The Russian Research Reactor Fuel Return (RRRFR) programme

The primary goal of the Russian Research Reactor Fuel Return (RRRFR) program is to advance nuclear non-proliferation objectives by eliminating stockpiles of Russian-origin highly enriched uranium (HEU). The program closely resembles the successful United States Foreign Research Reactor Spent Nuclear Fuel Acceptance program and works in conjunction with the Reduced Enrichment for Research and Test Reactors program [IAEA, 2009a].

The RRRFR programme was first conceived during trilateral discussions between the United States, Russia and the IAEA, initiated in 1999, when participants identified more than 20 research reactors in 17 countries having Russian/Soviet-supplied fuel. In 2000, the Director General of the IAEA sent a letter to 15 countries asking for their willingness to return HEU spent fuel to Russia. Fourteen countries responded positively to the Director General's letter.

In 2004, the Russian Federation and the US signed a Government-to-Government Agreement concerning cooperation to return the Russian-produced research reactor nuclear fuel to the Russian Federation. The United States, through the RRRFR programme, provides funding for the removal and reprocessing of eligible material. This agreement established the legal framework necessary for the cooperation between the Russian Federation and the United States for the return of Russian-supplied research reactor fuel from eligible countries. Under the Bratislava agreements concluded by Presidents George W. Bush and Vladimir Putin in February 2005, both countries committed to completing all shipments of Russian-origin HEU spent fuel currently stored outside research reactors by the end of 2010. By the end of 2009 the program had completed 22 shipments totalling over 910 kg of Russian-origin HEU spent and fresh fuel which has been returned from Bulgaria, the Czech Republic, Germany, Kazakhstan, Latvia, Libya, Poland, Romania, Serbia, Uzbekistan and Vietnam. During this time, the program successfully removed all HEU from three countries, Latvia, Bulgaria and Romania.

HEU spent fuel shipments have been the most complex shipments under the RRRFR programme. The first shipment of HEU spent fuel from Uzbekistan was completed in January 2006 followed by HEU spent fuel shipments from the Czech Republic in 2007, Latvia, Bulgaria, and Hungary in 2008, and Kazakhstan and Romania in 2009.

Contracts under the program include all of the preparation activities as well as the shipment, reprocessing, interim storage of the high level waste (HLW), and, in some cases, permanent disposition in the Russian Federation of the products of reprocessing.

6.8 Return of spent fuel from Serbia to the Russian Federation (RRRFR programme)

The largest and most demanding planned shipment of spent fuel within the RRRFR program is that of over 8000 fuel elements (corresponding to 2.5 tons of uranium metal) from the Vinca Institute of Nuclear Sciences in Serbia. This fuel was irradiated in the institute's RA research reactor, which operated from 1960 until 1984 [Bradley et al, 2007, Mukhatzhanova, 2008].

The RA reactor operated with TVR-S fuel elements, which consist of metallic uranium fuel clad in aluminium. The fuel is thus of the same type as the Norwegian JEEP I and early HBWR fuel. A temporary storage facility is located inside the reactor building. Fuel elements, about 17 percent of which contain highly enriched uranium (HEU), are stored mostly in water-filled aluminum barrels and steel containers. About 90 kg of irradiated fuel is kept inside the reactor itself. The lack of water circulation and chemical controls has damaged up to 30 percent of the fuel elements, allowing radioactive ^{137}Cs to leak into the water of the storage pool. Further leakages and physical degradation of the storage facilities could lead to a release of radiation into the environment, endangering Vinca employees as well as the surrounding countryside. Fuel degradation and corrosion pose a serious environmental risk and significantly complicate the task of removing and transporting fuel elements. Further challenges at this phase stem from the site's old and inadequate infrastructure, making it necessary to conduct updates and modifications to meet the project's technical requirements.

To deal with the situation, Serbian and international experts drafted a proposal to manage and secure the spent nuclear fuel and radioactive waste stored at the site. Along with basic safety concerns, more than 50 kg of unsecured fresh 80-percent HEU fuel presented a serious proliferation concern. After several months of complicated negotiations between Serbia, the United States, Russia, and the IAEA, the fresh HEU fuel was repatriated to Russia in 2002. In turn, the Serbian government requested the international community's help in dealing with the spent fuel and radioactive waste at Vinca. The effort to clean up the site constitutes the IAEA's largest technical cooperation program in terms of required funding (approximately 50 million US dollars), for which a number of IAEA member states, international organisations and non-governmental organisations provided the necessary funding. The program consists of three projects: removal and reprocessing (in Russia) of spent nuclear fuel; improved management of radioactive waste stored on site at Vinca; and decommissioning of the research reactor. The target completion date is 2015, while spent fuel repatriation to Russia should be completed by 2011.

Removing the spent fuel is the most expensive and complex project within the decommissioning program. In September 2006, Vinca and the IAEA contracted a Russian consortium of the Sosny Research and Development Company, Tekhsnabeksport, and Mayak Production Association to repackage the fuel and prepare it for transportation back to Russia. This work is currently underway, as are multilateral negotiations on the fuel's route. When the shipment from Vinca does take place, it will be the largest spent fuel repatriation to Russia to date.

6.9 Australia

Information in this subsection has been obtained from the Australian Nuclear Science and Technology Organisation (ANSTO) [ANSTO] and the French nuclear services company AREVA NC [AREVA NC].

ANSTO has operated three research reactors in Australia. The first, the High Flux Australian Reactor (HIFAR), operated from 1958 until January 2007. The very small Moata reactor operated from 1961 until 1998. Australia's newest research reactor, OPAL, first went critical in August 2006.

For most of its operating life, HIFAR used ^{235}U enriched to 60 per cent and clad in aluminium. The enrichment was eventually cut to less than 20 per cent. All HIFAR elements were manufactured in the UK, while the enriched uranium was supplied by both the UK and the USA. Over its lifetime, HIFAR has accumulated an inventory of more than 2000 spent fuel elements. Spent fuel from the HIFAR and OPAL reactors is securely stored for several years before being sent overseas for processing or storage. ANSTO, with the approval of the Australian Government, selected the overseas reprocessing path to treat the fuel for final disposal. The strategy involves the following successive steps:

- Reprocessing of spent fuel overseas, as no reprocessing facility has been established or will be established in Australia.
- Conditioning of the waste into a dedicated form (known as long-lived intermediate waste form), suitable for final storage.
- Return of the waste to Australia.
- Long-term storage of the conditioned waste in a national intermediate level radioactive waste storage facility.

Previously, Australian SNF was treated in the United Kingdom and the United States. The first shipment to the UK, in 1963, was of 150 elements, while the second, in 1996, was of 114 elements. The two shipments were sent to the United Kingdom Atomic Energy Authority's Dounreay plant in Scotland, where the fuel was reprocessed. HIFAR has also used US-origin fuel elements. In 1998, 240 spent fuel elements were shipped to the US Department of Energy's

Savannah River site under the Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Program. No wastes will be returned to Australia from these US-origin fuel elements.

Soon after this last shipment, the United Kingdom Government decided Dounreay would not enter into new fuel reprocessing contracts. Following negotiations, ANSTO and AREVA NC signed a contract in January, 1999. This contract includes the following services:

- Transport of the spent fuel elements in dedicated casks on specially equipped ships from Australia to France.
- Reprocessing of the spent fuel elements at the AREVA NC La Hague plant.
- Conditioning of the ultimate waste into a stable form.
- Return of ultimate residues for storage and final disposal in Australia.

The contract covers all non-US HIFAR spent fuel (about 1,300 fuel elements) and has provisions for the reprocessing of the spent fuel from the replacement reactor. The 1,300 HIFAR spent fuel elements will be transported in four shipments. The ultimate residues shall be returned to Australia, as provided for by the French law, and by the terms of the commercial agreement. Both the French and Australian Governments support the principle of the return.

The return shipments of residues, which will be encased in special canisters contained in dual-purpose casks (for transport and storage), will take place by 2015. The quantity of residues to be returned is estimated to be 36 canisters having a total volume of around 6 cubic meters.

The intermediate level waste resulting from reprocessing of HIFAR's fuel in France and Scotland, will be sent to Australia and stored in the proposed national radioactive waste facility. OPAL spent fuel will be sent to the USA for storage where it will remain.

6.10 India

Two research reactors (CIRUS and DHRUVA) are in operation at the Bhabha atomic research centre in Trombay, India [Ramesh, 2009]. Both employ metallic uranium fuel clad in aluminium. After discharge from the reactor, the fuel is placed in wet storage for approximately two years, and then reprocessed. Damaged fuel elements are put in aluminium cans and are reprocessed on shorter timescales.

6.11 Latin America

Several countries in Latin America operate research and/or commercial reactors, including Argentina, Brazil, Chile, Mexico and Peru. At the start of the current decade, these countries shared the view that it was essential and opportune to begin evaluation of options for interim storage and disposal of spent fuel, including any derivatives, if processing is one of the chosen options. This resulted in an IAEA Technical Cooperation (TC) Regional Project, “Management of spent fuel from research reactors in Latin America” [IAEA, 2006c].

Sixteen research reactors were included in the project; all of them were water cooled, either Material Testing Reactor (MTR) or Training Research Isotope Production General Atomic

(TRIGA) reactor types. The fuel types vary considerably, including UO_2 rods, plates of U_3O_8 dispersed in aluminium, U_3O_8 in polythene disks, UAl-Al dispersions and U_3Si_2-Al dispersions. The TRIGA fuel elements were U-Zr-H rods. Although most of these fuel types are not similar to the HBWR and Jeep fuels, most of them were clad in aluminium. The susceptibility of aluminium to degradation (that is, corrosion by water) lead to the conclusion that direct disposal cannot be considered a viable option for the fuel:

> *It is clearly understood that at some time in the future it will be necessary to process the spent fuel.*

For spent fuel processing, two main options can be considered: conditioning for disposal and reprocessing. Both produce long lived radioactive wastes that need to be disposed of in deep geological repositories. The main differences between them are the type and final volume of waste to be disposed.

As stated in its Radioactive Waste Management Strategic Plan, Argentina will decide by 2030 whether the spent fuel from its power plants is to be reprocessed. Research reactor fuel could also be reprocessed: this is a possible option because commercial services for reprocessing, including research reactor fuel, are offered in France. The other countries in the project have not established a date to decide whether reprocessing will be included as an option for spent fuel management. Reprocessing of all spent fuel produced in the research reactors of the Latin American region during a 30 year period would produce a total amount of 1.56 tons of HLW, representing 0.56 m^3 that can be accommodated in four COGEMA HLW canisters with a maximum capacity of 450 kg HLW each. In comparison, 220 m^3 of spent fuel derivatives would be produced if no reprocessing is done.

Conditioning produces low and intermediate level wastes (LILW), instead of the HLW that results from reprocessing, because the thermal loading of the spent fuel derivatives arising from conditioning is below 2 kW/m^3.

Currently, there are several different spent fuel conditioning technologies, at different levels of development. Some of these are partly or wholly concerned with dilution of ^{235}U, since many research reactor fuels (including most of the fuels in this project) have original enrichments in the range from 18 to 20 wt%. Excluding those concerned wholly with dilution, the technologies that were considered applicable in the regional context are:

- ***Can-in-canister***. Non processed fuel is placed in a can, which is placed in a canister into which is poured HLW glass to form a solidified unit.
- ***Melt and dilute***. The entire fuel assembly is melted and diluted with depleted uranium and aluminium (for eutectic formation). This technique was developed in the USA, and is discussed in Chapter 6.2.
- ***Plasma arc treatment***. The spent fuel is placed directly into a plasma centrifugal furnace with depleted uranium and neutron absorbers, where it is melted and converted into a HLW glass form.

- ***Glass material oxidation and dissolution system***. The spent fuel is placed in a glass melt furnace where it is oxidised by lead dioxide and then converted into a LILW glass form.
- ***Electrometallurgical treatment***. The spent fuel is melted with silicon and electro-refined. The bulk of the aluminium is electrolytically removed, and the residual aluminium, actinides and fission products are vitrified. Pure uranium is recovered.
- ***Chloride volatility techniques***. The spent fuel is reacted at high temperatures with chlorine gas and all the materials are converted to a volatile gas. The uranium, actinides and fission products are separated from each other by cooling and distillation.

It should be noted that the electrometallurgical treatment and chloride volatility techniques do not produce a disposable waste form, but are rather prior steps to either conditioning or reprocessing.

The above techniques can be divided into two groups: (i) those in which the final product will comprise an aluminium-based metallic matrix; and (ii) those with a glass or glass ceramic matrix. Researchers in the region involved in spent fuel R&D considered that the second group represents a better guarantee for long-term immobilisation of the radionuclides, as long as more evidence for long term behaviour is available.

In the Latin American region, conditioning processes are being studied on a laboratory scale for MTR fuels: TRIGA fuel conditioning has not yet been considered. Argentina currently has the capability to implement a selected conditioning process on a hot laboratory scale and to scale it up to pilot plant scale. After evaluation of the different conditioning processes, with and without decladding, one conditioning route will be chosen for testing in hot cells. Further, to avoid the need for isotopic dilution, an option that includes separation of uranium is under consideration. Under the Radioactive Waste Management Strategic Plan, several research lines are being studied. HALOX is a dry decladding process for aluminium cladding involving the chloride volatility technique, while a wet decladding process involves dissolution of fuel plates in 1M sodium hydroxide. Methods to immobilise spent fuel derivatives in glass or ceramic matrices include glass sintering, glass melting and ceramisation (CERUS process). The latter process consists of using a U_3O_8 powder to dilute and act as an immobilisation matrix for the spent fuel derivatives; the sintered U_3O_8 provides good corrosion resistance.

These various techniques are being considered as components of integrated processes, such as:

- Dry decladding followed by dissolution/isotopic dilution and vitrification or ceramisation.
- Wet decladding followed by dissolution/isotopic dilution and vitrification or ceramisation.
- Complete wet dissolution followed by dissolution/isotopic dilution and vitrification or ceramisation.
- Direct ceramisation of spent fuel plates (CERUS process).

Operational start-up of a deep geological repository is currently planned for around 2050.

7 STORAGE OF SPENT FUEL AND CONDITIONING PRODUCTS

7.1 Storage options

A brief overview of the historical development of spent fuel storage facilities was presented in a recent IAEA report [IAEA, 2009b].

In the early period of nuclear development, the pools for storing spent fuel at the plant were built with small capacity for only a few years' storage, on the assumption that spent fuel would be shipped to a reprocessing plant for recovery and reuse of contained uranium and plutonium. However, since reprocessing has in many cases not been the main spent fuel management strategy, the need arose for new storage technologies, especially of dry types, in addition to the classical method of storage in water pools. A historical evolution of storage technologies is summarized in Table 7-1.

Table 7-1. Historical evolution and deployment of spent fuel storage systems

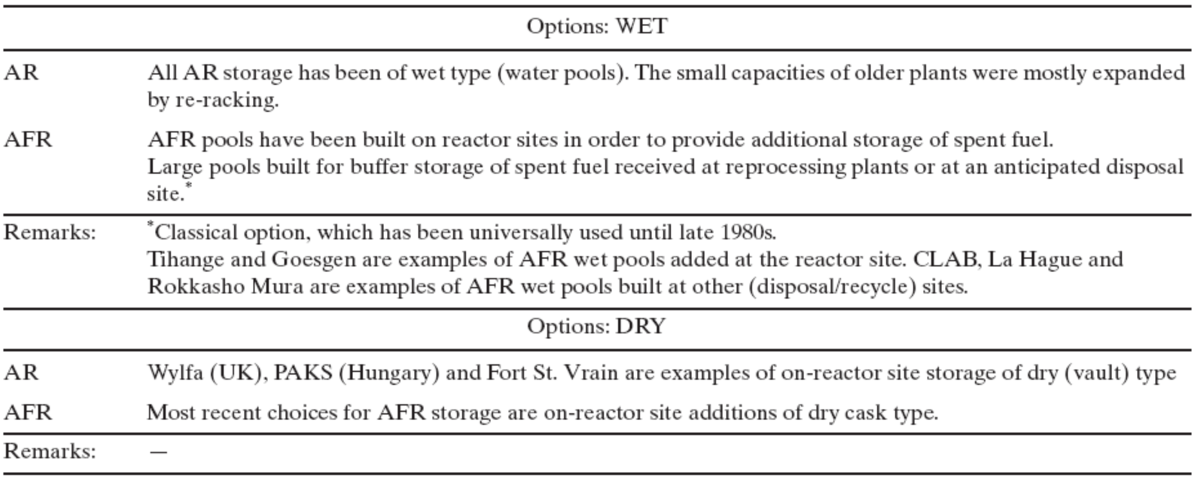
Option	Year					
	1950	1960	1970	1980	1990	2000
WET	Most of the AR and AFR pools					
DRY			• Vault (1971, Wylfa)			
			• Concrete Silo (1977, Whiteshell)			
				• Metal Casks (1986, Surry)		
				• Gorleben (CASTOR)		
					• Concrete Casks (1992, Surry)	

There are several aspects that are used for categorizing spent fuel storage facilities:

- At reactor (AR) versus away from reactor (AFR);
- AFR: on-reactor site versus off-reactor site;
- Wet versus dry.

Some general remarks about these different aspects are contained in Table 7-2.

Table 7-2. Alternatives for spent fuel storage



	Options: WET
AR	All AR storage has been of wet type (water pools). The small capacities of older plants were mostly expanded by re-racking.
AFR	AFR pools have been built on reactor sites in order to provide additional storage of spent fuel. Large pools built for buffer storage of spent fuel received at reprocessing plants or at an anticipated disposal site.*
Remarks:	*Classical option, which has been universally used until late 1980s. Tihange and Goesgen are examples of AFR wet pools added at the reactor site. CLAB, La Hague and Rokkasho Mura are examples of AFR wet pools built at other (disposal/recycle) sites.
	Options: DRY
AR	Wylfa (UK), PAKS (Hungary) and Fort St. Vrain are examples of on-reactor site storage of dry (vault) type
AFR	Most recent choices for AFR storage are on-reactor site additions of dry cask type.
Remarks:	—

Today, most new storage is AFR. But the question remains as to whether or not to construct these facilities on reactor sites or at a new site (this issue is being addressed by the Phase 2 Committee).

The technologies currently available for spent fuel storage fall broadly into two categories, wet and dry, distinguished according to the cooling medium used (Table 7-3). The wet option has historically been used for temporary storage and cooling at reactor sites and in some interim off-site storage facilities generally associated with disposal or reprocessing sites (in anticipation of the next step in the cycle). More recently, however, a variety of dry storage options have been developed and applied in the international market. There are several dry storage designs available from vendors in the international market that differ in design details. There are four generic types of dry storage:

- Metal cask.
- Concrete cask/module.
- Vault.
- Other (drywells, tunnels, etc.).

The technologies are distinguishable by their major technical characteristics, namely: the predominant heat transfer method; type of shielding; transportability; location with respect to the geological surface; degree of independence of the individual storage units; and the storage structure. They also differ in terms of materials of construction, size, modularity, spent fuel configuration, layout of the storage containers (horizontal, vertical, etc.) and methods for fuel handling.

Table 7-3. Technical options and applications for spent fuel storage

Type	Option	Heat transfer	Containment (medium)	Shielding	Feature	Examples
Wet	Pool	Water	Water/building	Water	Classic option	Most ARs & many AFRs worldwide
Dry	Metal cask	Conduction through cask wall	Double lid Metal gasket (Inert gas)	Metallic wall	Dual purpose	CASTOR, TN, NAC-ST/STC , BGN Solutions
	Concrete cask/silo	Air convection around canister	Cavity lining/ seal welding (Inert gas)	Concrete and steel overpack	Vertical	CONSTOR, HI-STORM/HI-STAR
	Concrete module	Air convection around canister	Canister sealing (Inert gas)	Concrete wall	Horizontal	NUHOMS NAC-MPC/UMS MAGNASTOR
	Vault	Air convection around thimble tube	Thimble tube (Inert gas)	Concrete wall	Several cases	MVDS MACSTOR
	Drywell/tunnel	Heat conduction through earth	Canister (Inert gas)	Earth	Below ground	Not commercialized

For future storage of spent fuel and/or conditioning products in Norway, the committee has assumed that dry methods will be employed.

As reported in Chapter 6.4, one of the contingency options being investigated in the UK for spent Magnox fuel in the event that reprocessing facilities are not available is dry storage. A summary of relevant store types and designs was given in a recent paper [Morris et al, 2009]. In particular, the study investigated vaults, casks and silos.

A vault (Figure 7-1) is a concrete structure containing storage pits or wells, with shielding being provided by the building structure. The spent fuel is stored in tubes, and heat is removed by convection of air across the external surface of the well. The SNF can be either bare or placed in containers.

Examples of vault storage facilities include the CASCAD (CASemate CADarache) facility in France, the Paks MVDS facility in Hungary, the Idaho spent fuel facility in the US and the Chalk River facility in Canada. The CASCAD facility is used to store spent fuel from CEA research reactors, French naval reactors and UNGG fuel from early gas-cooled reactors. The fuel is stored in helium filled canisters in vertical wells (Figure 7-2).

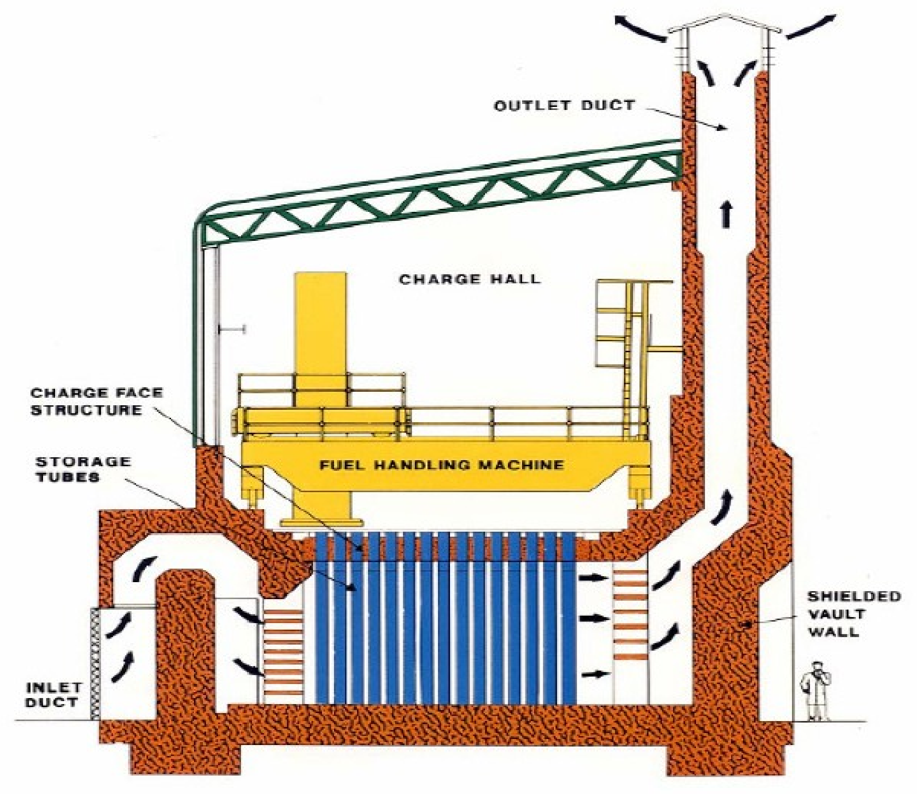


Figure 7-1. Vault store design

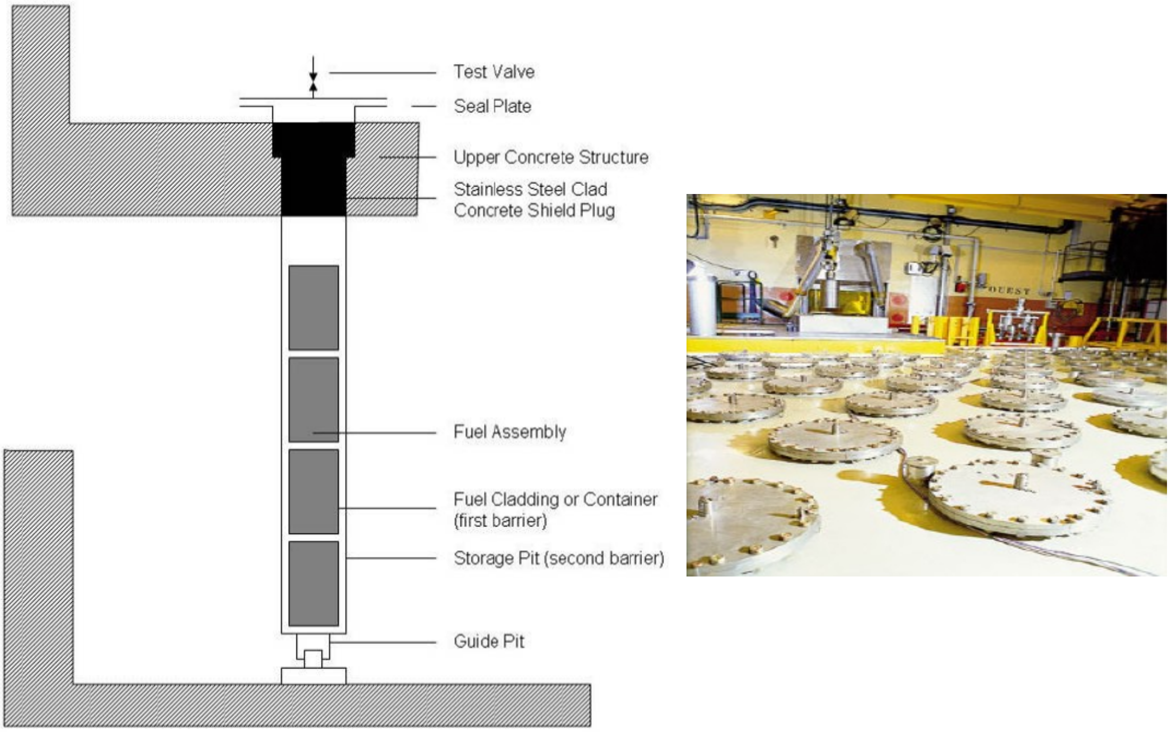


Figure 7-2. CASCAD vault storage system

The facility at Chalk River (Figure 7-3) will be used to store legacy Al-clad uranium metal fuels in vented canisters, allowing hydrogen gas to be vented and preventing formation of uranium hydride. The fuel will be retrieved from current tile-hole storage and vacuum dried.

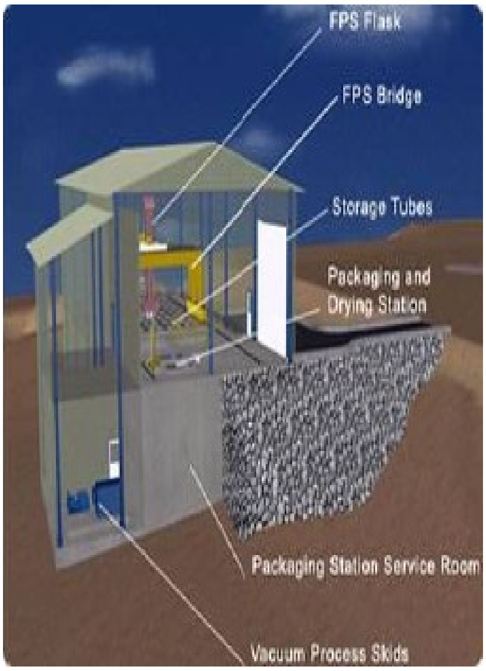


Figure 7-3. Chalk River vault storage system

The HABOG facility in the Netherlands is a multi-function dry storage vault that stores spent MTR fuel from the research reactors at Petten and Delft as well as high active waste from reprocessing of spent fuel from the commercial power plant operations at Borssele and Dodewaard [Kastelein, 2005]. The facility is a reinforced aircraft-crash and flood resistant structure that has been designed for fifteen years of active use and 85 years of passive operation.

HABOG contains two separate compartments: one for storage of non-heat generating high-level waste from the reprocessing of spent power reactor fuel, such as fuel element cladding, hulls and ends, the other for storage of the heat-generating, vitrified HLW from reprocessing of spent fuel from the NPPs, and of un-reprocessed spent fuel from the research reactors (Figure 7-4). The canisters of heat generating waste are stacked in vertical, air-cooled storage wells, which are filled with argon to prevent corrosion of the canisters and are equipped with a double jacket to enable air-cooling through natural convection while avoiding direct contact of the cooling air with spent fuel or vitrified HLW canisters. Upon receipt at HABOG, the research reactor spent fuel is repacked in steel canisters, which are then filled with helium to prevent chemical deterioration of the spent fuel during long-term storage into account in the design of the facility.

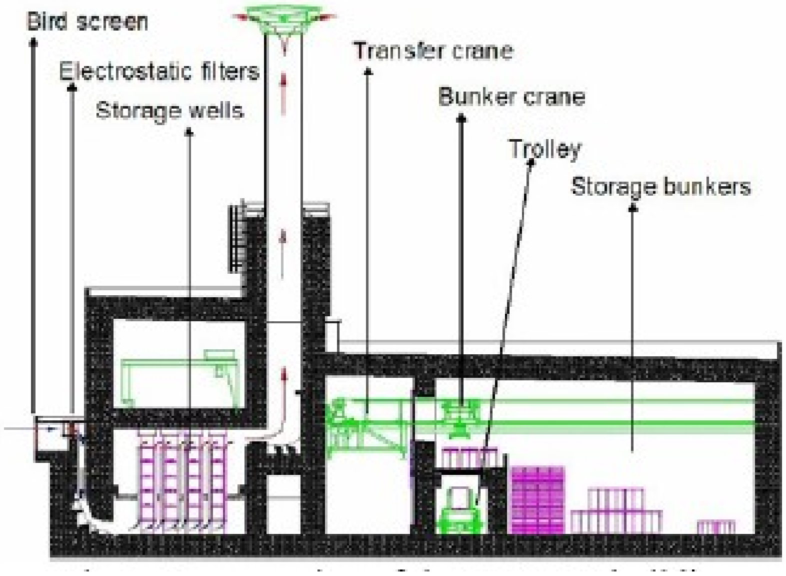


Figure 7-4. HABOG storage facility

The HABOG project was initiated in 1994 and commissioned at the end of 2003 at an estimated cost of $100 million. This total includes ~$75 million for design, licensing, site development and construction of the major receiving facilities and the hot cell and $5 million each for the five associated storage areas.

Another major storage facility for research reactor fuel, the Idaho Spent Fuel Project (ISF), is under development at Idaho Falls in the United States [Roberts et al, 2003]. This is planned to hold the spent fuel accepted under the FRR SNF programme, spent TRIGA fuel, and spent fuel from the Shippingport and Peach Bottom power reactors. The facility has been designed for 50 years of operation and is based on the Alstec modular vault dry storage (MVDS) technology. Additional storage modules can be added as needed.

The interior of the ISF is shown in Figure 7-5. The facility consists of three main functional areas: the Cask Receipt Area, the Transfer Area and the Fuel Storage Area. Fuel is accepted into the Cask Receipt Area, then repackaged into canisters in the Transfer Area, and then placed into storage in the Storage Area. The ISF facility uses the DOE standardized spent fuel storage canisters that are intended to be compatible with a future high level waste repository The ISF is designed to easily retrieve the canisters from storage for shipment to the geologic repository when this becomes available.

The originally quoted cost of this storage facility was $137 million according to a competitively bid contract awarded to Foster Wheeler Environmental Corporation (FWEC). By comparison, the total investment for a 10 module dry storage vault constructed at a reactor site is estimated at around $58 million, because advantage can be taken of shared infrastructure.

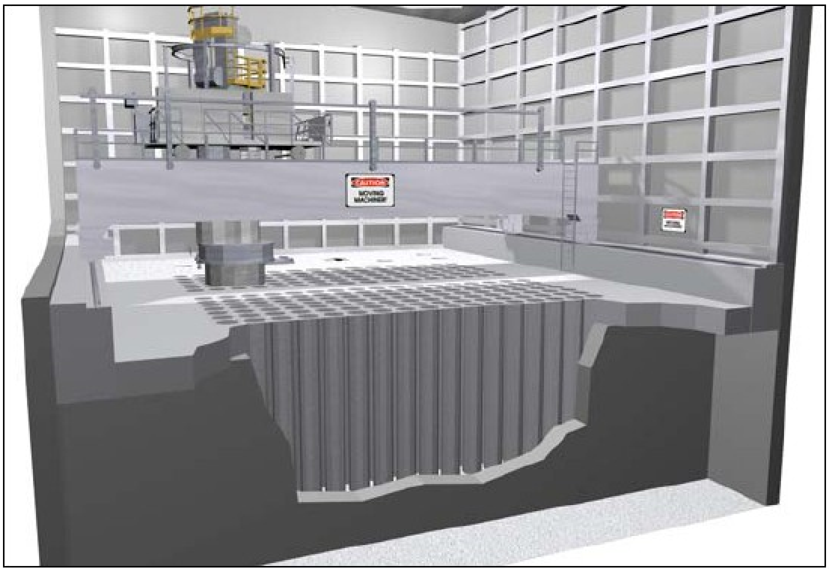


Figure 7-5. The proposed Idaho spent fuel project storage vault

Canisters of vitrified high level waste are often stored in concrete vaults. An example is given from a design concept for storing canisters filled with vitrified waste produced at the Defense Waste Processing Facility (DWPF) at the Savannah River Site [Rainisch, 1993].

The main building is a reinforced concrete structure that includes a storage vault; a loading and unloading area; and air inlet and outlet shafts. The vault area is a partially below-grade reinforced concrete structure. The vault floor is located 12 ft below grade so that the floor in the adjacent loading area is at grade level. Inside dimensions of the vault are 173.5 ft L x 51 ft W x 46 ft H. To satisfy shielding requirements, the vault's concrete walls are 3 ft 6 in. thick and the roof slab is 3 ft thick. The selected shielding would reduce outside dose rates adjacent to the vault's wall to approximately 1 mR/hour. The vault is subdivided into three compartments, and each compartment would be equipped with a rack system for locating and supporting the canisters. Air intake and outlet shafts constructed of reinforced concrete would provide the vault area with cooling air. Each compartment has a rack system for storing the glass canisters. The glass cylinders stand in their upright position on a centre-to-centre spacing of 30 in. Because the canisters are 24 in. in diameter, 6 in. remains between canisters to accommodate the structural grid that accepts and supports the canisters. The 6-inch spacing will also be used for convective air currents for cooling the canisters. A 20 ton overhead bridge crane is housed within the structure and is used to store and retrieve canisters. A mechanically induced air cooling system serving the vault area will remove decay heat and will ensure that the vault's air temperature remains within specified limits.

Similar structures are used for storage of HLW at the Sellafield and Cap La Hague reprocessing plants (Figure 7-6).

Figure 7-6. Vitrified waste canister and storage facility at Cap La Hague

In cask storage, a moveable concrete or metal structure provides containment and shielding. There is no requirement for a shielded building, although a covered facility may be preferred. Heat is removed by convection or conduction, depending on materials and cask design. Some casks can be licensed for storage and transport, and potentially for disposal.

The CASTOR-MTR 2 cask from GNB in Germany, Figure 7-7, has been developed to accept spent fuel from the German research reactors. The cask is designed for both transportation and up to 40 years of interim storage of the fuel. The cask body is made of ductile cast iron and weighs about 18 metric tons. The CASTOR MTR 2 has both primary and secondary lids with the interstitial space filled with helium. The helium gas pressure is continuously monitored during storage to confirm the leak-tightness of the cask. The cask can accept several different types of inner basket, depending on the type of fuel to be stored.

Figure 7-7. CASTOR-MTR 2 cask

The TranStor concrete cask (Figure 7-8) has been developed by BNFL in the US. It is designed for PWR and BWR spent fuel, for both transport and storage. The fuel is placed in a carbon steel basket underwater in the reactor pond, and a welded stainless steel canister provides containment. A transport overpack (stainless steel and lead) provides shielding during loading and transfer operations. On arrival at the storage location, the canister is transferred to the reinforced concrete storage overpack, which is vented for cooling and provides protection and shielding. The overpacks can be stored in the open air. These casks have been used to store metallic uranium legacy spent fuel from INEEL.

Figure 7-8. TranStor casks

The Canadian MACSTOR concrete casks (Figure 7-9) are used to store CANDU spent fuel, which is first placed in galvanised carbon steel canisters. Each MACSTOR module, which are stored in the open air, holds 20 CANDU fuel canisters or two LWR fuel canisters.

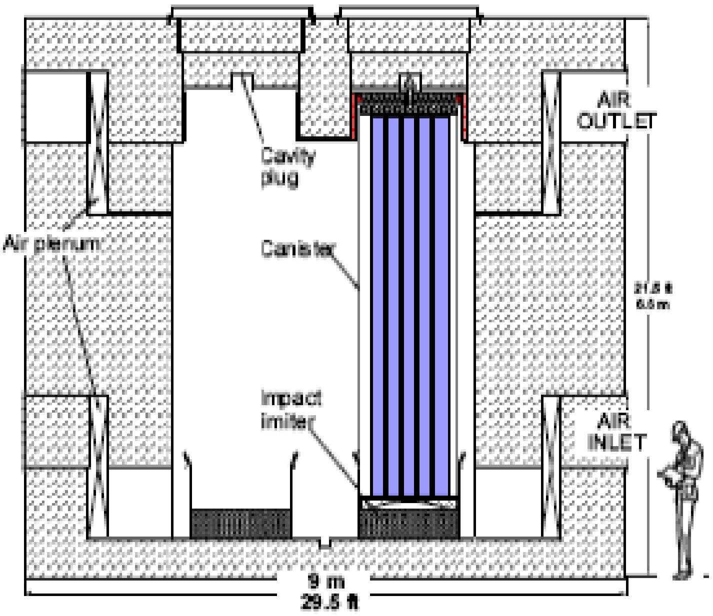


Figure 7-9. MACSTOR concrete casks

Silos are monolithic or modular immovable concrete structures, which provide containment and shielding. A reinforced concrete canister, with a welded steel inner liner, has been constructed by AECL for CANDU fuel (Figure 7-10).

Figure 7-10. AECL concrete silos

The NUHOMS silo storage system (Figure 7-11) uses sealed metal canisters to contain the spent fuel. Fuel is loaded vertically into the sealed metal canisters, which are stored in a horizontal orientation inside the concrete storage module. An on-site transfer cask is used to transfer the fuel bearing canister to the horizontal concrete storage module.

Figure 7-11. NUHOMS storage system

The current status of management practices in several countries for the interim storage of research reactor spent nuclear fuel (RRSNF) was summarised in an IAEA Technical Meeting, *Guidelines of Good Practice for the Management and Storage of Research Reactor Spent Fuel,* in October 2009 [IAEA, 2009c]. The purpose of the meeting was to gather information which will be used in a guidance document covering interim storage of research reactor spent nuclear fuel, and which will include examples of best practices and lessons learned.

The meeting included discussion, among others, of the following topics:

- National legislation, institutional and political aspects and public acceptance.
- Sources and quantities of RRSNF.
- National practices, facilities and capabilities.
- National plans/options related to storage of RRSNF for the next 50 years.
- Comparison between wet and dry storage of RRSNF.

The meeting demonstrated the diversity of practices with respect to interim storage of spent fuel at research reactor facilities. Many research reactors still use wet storage, although there is an increasing tendency towards dry storage. It was revealed that many Member States are currently depending on the U.S. or Russia for the eventual disposition of the fuel under the existing take-back programs. Very few Member States have the flexibility or the infrastructure to support long term disposition of spent fuel.

The examples of national practices in the following subsections are taken from presentations at the above meeting. Emphasis is placed on dry storage as this is considered to be the most likely option for the future in Norway (that is, for HBWR oxide fuels and conditioning products from processing of metallic and JEEP II fuels).

USA

Within the United States, the Department of Energy (DOE) is responsible for management of legacy fuels from defence programs and all fuels from research reactors. Included in this inventory is legacy material from commercial power reactor development and core debris from the Three Mile Island reactor accident. All of this material is managed at three principal interim storage sites, Hanford, Idaho, and Savannah River. For good long term dry storage, important issues include retrieval and dewatering, packaging for extended storage, and characterization sufficient to support drying.

Currently funded R&D programs aimed at addressing dry storage issues (including neutron absorber development, canister welding and canister development) are essentially complete. Neutron absorbers will allow for packaging DOE fuels in anticipated disposal configurations to avoid very restrictive fissile loading limits for disposal. Canister welding supports packaging without use of shield plugs, lowers worker exposure, and decreases the number of disposal packages. All SNF can be dried stored if suitable designs are used, to allow retrievability and prevent additional fuel degradation.

Wet and dry storage criteria are conditions/practices needed to enable full retrievability of spent nuclear fuel following its targeted storage period. At present, the criteria are essentially embedded in various practices and guides. Acceptance Criteria for dry storage include an assessment of degradation mechanisms in aluminum-based SNF in humid environments, including corrosion and creep, thermal models for dry storage and the development of empirical models to establish temperature limits and acceptance criteria for dry storage systems.

Under the Foreign Research Reactor SNF Acceptance Programme, approximately 9000 fuel assemblies have been received and are in storage at Idaho (non-aluminium based fuels) and Savannah River (Al based fuel): the aim is to group together like fuel types to improve future fuel management. There will be a major campaign of fuel moves (~300 shipments) over 7-10 years to transfer all Al based fuel currently at Idaho to Savannah River and all non-Al fuel from Savannah River to Idaho. There is a compliance agreement between Idaho and USDOE to move all fuel from wet to dry storage by 2023. A repackaging plant will be constructed at Idaho to place all non-Al fuel into standardised canisters. All repackaging of non-aluminium fuel will be completed at the new Idaho plant. The current strategy is to process the Al fuel: this may start in 2011 depending on outcome of environmental review and analysis.

Canada

The ultimate aim of Canadian national policy is to have a centralized geological repository. Prior to operation of this planned repository, Canada's interim storage strategy involves continued wet/dry storage at reactor sites. The final disposal solution, i.e., geological repository, is being developed in parallel.

Currently, there are eight operational and six shutdown research reactors in Canada. There is a large inventory of RRSNF associated with the operation of these reactors. There is also large variation in the fuel types and configurations in wet or dry storage across the reactor sites. Canada is evaluating safety issues associated with current interim storage strategy including potential hazards and design considerations and is currently implementing improved dry storage systems. The Canadian strategy involves the retrieval, conditioning, and dry-storing legacy fuels into a new above ground dry storage facility. An additional dry storage system will be constructed for future spent fuel generation from NRU and other types of radioactive wastes generated at the Chalk River site.

The spent fuel from Canadian research reactors has been initially stored underwater for a few years before being dry stored in dry wells (tile holes) at the Chalk River Waste Management site, and in concrete canisters at the Whiteshell site. The spent fuel from these research reactors has various types and configurations, typically clad in aluminium with core materials of uranium metal, aluminium-uranium alloys and uranium silicide dispersed in aluminium.

Routine evaluation of first generation tile holes and their contents gave indications of water and salt ingress into the tile holes, with subsequent contact with the spent fuel cans contained within. These conditions are caused by the design of the tile holes, which is conducive to degradation of the spent fuel canisters that have been stored in the tile holes for 30-40 years. A newer design of the tile holes improved the conditions considerably. In addition, a project was initiated to prepare the fuel cans for retrieval of the breached cans and recovery/stabilization of the contents. This includes puncturing the cans to relieve gas pressure, fuel retrieval, drying, and repackaging.

A new above ground dry Fuel Packaging and Storage Facility (FPS) will accommodate fuel retrieved from the First Generation tile holes and is due to be built by 2011. The facility is designed to deal with all research reactor fuel types and repack them according to repository requirements. Inspections of Second Generation tile holes revealed little or no deterioration of fuel cans. This technology, although successful, will be phased out from 2015.

A new modular dry storage facility (NDSS) is to be constructed to accommodate future spent fuel generation at Chalk River (after initial pond cooling for on average two years). The first module will have a 10 year capacity. The facility will be under cover for better operational flexibility. Both the FPS and NDSS will have monitored vented storage and will be licensed to operate for 50 years.

The cost of the FPS with 100 tile holes will be approximately $82M based upon assumptions of affordable stabilization of the degraded fuel. The modular NDSS will cost approximately $155M for delivery of a 10 year capacity module excluding operating costs. The second generation tile holes contain 1000 HEU assemblies and 2000 LEU silicide assemblies that are eligible for return through the U.S. foreign returns program. Funding is being considered for a feasibility study to retrieve these and advance the date of retrieval from 2050 to 2019. The roughly 38 t U of very low burnup uranium metal fuel currently hanging in the reactor hall have indications of hydride blister formations. These assemblies were desheathed and resheathed in 2000 and will be used to test the new drying facility.

Hungary

The Budapest Research Reactor (BRR) has been in operation since 1959, and operates approximately 3500 hours per year. BRR has operated on HEU fuel and recently (September 2009) commenced conversion to LEU fuel. The reactor has produced a total of 2,070 spent fuel elements since 1959, and in 2008 a first spent fuel shipment was made under the Russian return program. A second shipment is scheduled for 2014. BRR has 2 wet storage pools for spent fuel.

The ability to encapsulate spent fuel assemblies (SFAs) and store them in an AFR-pool provided BRR with extended storage capacity. With the design of the AR-pool it is difficult to inspect the SFAs, and a Nuclear Underwater Telescope (NUT) was constructed to perform visual inspections of the SFAs. In 2001 BRR decide to convert to semi-dry storage of their SFAs to overcome the deficiencies of long term (40 years) wet storage. BRR designed an encapsulation can which provided sealed tube storage (drying, vacuuming, filling with inert gas, closing hermetically). This provided an intermediate storage option for +50 years, a solution for both fuel types, easy monitoring after canning and left open the option for a final long term solution. The fuel encapsulation technology involved the placement of each fuel assembly in a hermetically sealed, vacuum-dried, and back-filled aluminium container, 3 mm thick and nearly 1 m long canister with an Al-clad stainless steel bottom weight. BRR encapsulated 342 SFAs into the new cans prior to halting canning due to the start of the Russian Take Back Program. The encapsulated assemblies were subsequently opened for return to Russia.

Latin America

Activities in Brazil and Latin America are related to the dual purpose cask for dry storage of spent research reactor fuel in the region. The first phase of the project was carried out by the five Latin American countries involved in a study to select the best long term management option for SNF. They identified the need to develop an alternative solution to the U.S. returns program.

The second phase was to develop a storage cask which would be dual purpose for long term storage and transport of SNF. The plan is to use the cask in the region but license it for use elsewhere. The cask is expected to be used for storage for up to 50 years but could also be used now for fuel transports. None of the countries involved in developing the cask are under obligation to use only this cask type.

Australia

Thirty five HIFAR fuel assemblies are removed per year. The fuel is stored for 3 - 4 months at reactor and up to three years in the interim storage pond. The fuel will then be shipped to the US. The interim storage pond has capacity for 240 assemblies in racks.

ANSTO has a facility that is available for dry storage of spent fuel. The dry storage facility (DSF) was built in 1968, with 50 x 16 m holes with stainless steel liners. The capacity is 110 SFAs and the tubes are filled with nitrogen gas. ANSTO uses a 40-year old general purpose flask for the transfer of fuel from the wet to the dry storage facility. All spent fuel was removed from dry storage in 2007, and the currently empty facility is expected to be used for tails during the Mo-99 program. The wells in the DSF are sealed compartments with relative humidity less than 20%. Any reading above 50% is an indication of a poor seal. The cells are monitored for leaking as indicated by increased relative humidity or oxygen.

Nineteen elements of the 2281 that were fabricated showed extensive pitting corrosion at the tube ends and along the electron beam welds that join the three curved plates into a cylinder. A total of 2281 fuel assemblies have been shipped from ANSTO for conditioning. The OPAL reactor operation creates 35 SNF assemblies per year. The fuel is stored in the service pool, which has 10 years of available capacity. Spent nuclear fuel can be returned to US before 2016 deadline or to Argentina. Post 2016 irradiated SNF will be sent for reprocessing to Cogema or Argentina.

Dounreay, UK

The current inventory at Dounreay includes 110 MT of fuel, with DFR elements making up the largest quantity at 44 MT. There is a reprocessing option only for the 44 MT of DFR fuel. Three standard containers are being considered for the packaging of the remaining fuels. The canisters would be stored on an interim basis in a to-be-built storage facility on site. The container is designed for 75 years storage, and is constructed from 316L SS with a welded lid with vacuum and gas purge. The fuel will be free from moisture, halogen and sodium residue.

Nuclear Research Institute Rez plc (NRI), Czech Republic

All fuel elements used in the LVR-15 reactor from the start of the reactor operation in 1957 until 2007 had been stored in storage facilities at Rez. A new transport cask system VPVR/M ŠKODA has been designed to transport RRSNF.

An at-reactor pond (AR pond) is located in the reactor hall and serves as an interim spent fuel storage as well as for handling fuel from the reactor core during reactor outages or under off-load refuelling. The capacity of the AR pond is 60 places for fuel assemblies, 30 of which are held empty for emergency fuel movements from the reactor core. In addition, there is the reactor annex pond with a total capacity of 100 assemblies located on the reactor site near the reactor hall. The HLWS facility also includes a handling hall equipped with a crane and an overhead handling facility. The hall serves for preparation of containers before transfer to the hot cell.

Future plans involve the conversion to LEU fuel over the next two calendar years. A second shipment of HEU to Russia is planned in 2014. The LVR-15 is expected to operate until 2028. Plans are to provide wet storage of SNF for 3 - 5 years at AR and AFR-1 facilities followed by dry storage in ŠKODA VPVR casks for up to 50 years. No decision has been made on a final disposition option. Reprocessing and a permanent repository are both being considered.

South Africa

In South Africa, spent fuel from the SAFARI-1 research reactor is stored in the Thabana Pipestore dry storage facility after an initial cooling period of at least two years in the reactor pool [IAEA, 2009d]. The Pipestore comprises subsurface sealed stainless steel storage pipes positioned in boreholes that are lined with normal borehole lining and cement pipe for possible acid neutralisation. The pipe openings are sealed with lead plugs and an airtight flange, and filled with an inert gas. The subsurface Pipestore/borehole design was selected to provide both radiation shielding and heat transfer.

7.2 Costs

For new spent fuel storage facilities, various options can be considered on the basis of capacity needs, technical criteria and perceived contingencies. It is mainly a matter of defining the objectives and making a good selection from the available options. Today, selection and acquisition of storage facilities can be achieved through the tendering process among several competing technologies and suppliers. The tendering process, including development of relevant scope and appropriate requirements and criteria, is perhaps the most critical step in the process.

There are several technologies with reasonable availability for dealing with the problem of long-term storage of research reactor spent fuel [IAEA, 2009b]. When reprocessing is selected as the preferred route for spent fuel management, provision for long-term storage of vitrified HLW is likely to be required, pending final geological disposal.

Taking all relevant factors into account, it is likely that an independent facility would require an initial investment in the order of $100 million and as much as $1 million/year to operate and keep secure. Regional facilities would have the advantage of sharing or distributing the cost but introduce some new cost and political issues.

The regulatory, infrastructure and fixed construction costs are the major expenses, which are quite independent of the nature or the amount of material being managed.

8 DISCUSSION

8.1 Spent fuel conditioning

There are many issues involved in the selection of a spent fuels management strategy, and thus a number of criteria are required against which to assess the merits of various options. Examples were given in previous chapters outlining the decision processes in the case of the R1 reactor fuel in Sweden, in the USA concerning N Reactor fuel, and in the UK with regard to Magnox and DFR breeder fuels should the preferred strategies not be available.

Each country possessing spent fuel will have to make a decision on its management based on a unique set of criteria, although many of these criteria will be common. The final decision will depend on, among others, political aspects, geography/geology, and the availability of technology, resources or access to domestic and/or foreign services. Any option will likely have associated advantages and disadvantages which must be carefully weighed against each other. In some cases, the advantages and disadvantages will be finely balanced, while, in others there can be a "show-stopper" that can rule out the option whatever its advantages otherwise may be.

As an example, in Canada strategies are assessed against the following criteria [AECL, 2009]:

1. Socially acceptable.
2. Technically sound.
3. Environmentally responsible.
4. Economically feasible.

In the UK, similar criteria are being used [DSRL, 2009]: preferred processes should use mature technology and be:

1. Safe and secure.
2. Environmentally responsible.
3. Value for money.
4. Publicly acceptable.

In order to assess the various options that may be considered for the Norwegian metallic uranium and aluminium fuels, the committee has chosen criteria based on those listed above, but also including additional criteria specifically concerned with international experience and recommendations. The criteria are:

- The recommendations of the IAEA and OECD NEA, including ethical aspects.
- Technical suitability.
- The maturity and availability of the technology.
- The decisions taken by other countries on similar fuel types.
- Environmental impact.
- Value for money.

- Public acceptance.

In general, the issue of public acceptance has not used to compare the conditioning options since the committee members do not consider themselves competent in this area, and since such an analysis could readily result in all possible technical options being discounted, as in each case at least one stakeholder could be identified who would likely object to the proposed solution.

From the treatment options that either have been used or have been considered for commercial and research reactor fuels (discussed in Chapters 5 and 6) the following have been considered for the Norwegian metallic uranium and aluminium fuels:

1. Direct disposal or “once through fuel cycle”.
2. Long term interim storage with postponed decision or “wait and see option”.
3. Exchange of fuel.
4. Return to the country of origin.
5. Commercial processing using the PUREX technique.
6. Conditioning in Norway.
   A. Conditioning using developments of PUREX.
   B. Electrometallurgical conditioning (pyroprocessing).
   C. Calcination.

These options are discussed individually in the subsections below, and are compared against the selection criteria in Table 8-1.

8.1.1 Direct disposal

Direct disposal would have the advantage that a conditioning step, which would itself result in production of radioactive wastes, would not be necessary. In addition, the costs of this option would be favorable compared with those that do necessitate conditioning. The concept of direct disposal is widely accepted for oxide fuels clad in Zircaloy.

However, metallic uranium is potentially pyrophoric, and reacts with water to form hydrogen gas and uranium hydride, which is itself pyrophoric. In addition to the risk of explosion, the formation of gaseous reaction products could increase the pressure in any storage or disposal containers. Aluminium cladding can undergo severe localised corrosion under certain conditions, which can result in penetration of the cladding.

There is general international consensus that metallic uranium fuel and/or fuel clad in aluminium are unsuitable for direct disposal because of the above physical and chemical properties. Examples in which direct disposal has been formally rejected include the Swedish R1 reactor fuel and aluminum spent fuel at the Savannah River Site in the USA (see Chapter 6).

The committee considers that for the type of fuel under consideration, direct disposal does not meet the requirement that the fuel management strategy should be environmentally sound.

8.1.2 Long term interim storage with postponed decision

As reported in Chapter 5.2, spent fuel is stored to satisfy one or more of several waste management needs, the more important of which include:

1. The development and maintenance of a materials backlog; for example, to ensure the continuity of process operations.
2. The provision of hold-up capacity to allow radioactive decay, thus reducing radiation and heat generation levels in the waste.
3. The accumulation of sufficient quantities of waste to enable economic treatment, transport, recovery etc.
4. The provision of hold-up capacity while optimum methods are evaluated for treatment, further storage and disposal.
5. The storage of materials that, while not immediately required, have some potential future use or value, and have therefore not been defined as waste.

If the current situation in Norway is considered, it is clear that neither points 1, 2, 3 or 5 provide justification for long term storage with postponed decision on the final disposal route. Point 4 is strongly linked to the following statement from the NEA RWMC:

> *If a policy of long term storage is adopted in view of possible future developments, for example solutions based on technology development or multi-national facilities, then the country adopting that policy must work actively to investigate and develop these possible solutions. Waiting, on its own, is not enough.*

Since there are no plans (that the committee is aware of) to develop a domestic treatment process for spent fuels in Norway, the “wait and see” option cannot be justified. In many countries in which “wait and see” is being considered, there are either outline plans which assume that future domestic reactors/technology can be used to ”burn” fuel that is currently stored, or there are programs in operation to develop treatment methods.

Some stakeholders may hold the view that, even if a domestic technology is not under development, there are circumstances in which storage and postponed decision may be justifiable. These circumstances may be related to expectations that improved processes will become available elsewhere.

In such a case, a definition of what constitutes an improved process is necessary. Based on the objections to the currently available process (reprocessing using the PUREX process), it may be postulated that an “improved” process may be defined as one that (i) would not produce a pure plutonium product stream, and would hence offer improved proliferation resistance, and (ii) would produce lower radioactive discharges to the environment.

Such a decision to pursue storage and postponed decision should not be taken blindly, and consideration should be given to the prospects of a more-favourable treatment being developed. Such consideration should take into account the rational for new developments: treatment of SNF requires extensive resources and political support. Such developments of new spent fuel treatments are thus expected to be linked to national plans for commercial nuclear power: the developments in reprocessing and pyroprocessing discussed in Chapter 5.3 are all related to Generation III+ or Generation IV reactor types.

Since it will take several decades at least until any of these new processes are in regular operation, it is difficult to envisage development of any other new techniques for spent fuel treatment. Further, there is no obvious reason why any alternative treatment would be developed for metallic uranium and/or aluminium-clad fuels, since the PUREX method is demonstrably suitable for these fuels.

For all the alternatives to the current treatment methods must be measured several other considerations. Firstly, it is not known whether any such plants will actually be constructed. Secondly, should they be built, it is not known when operation would commence. Finally, if these replacement facilities are built, it is not known whether commercial services will be offered.

In addition to these ethical-based objections to interim storage with postponed decision, there are also several more practical considerations. Spent fuel will, at some point in time, degrade. Damaged and/or corroded fuel is more difficult, and thus expensive, to handle. The degraded Swedish R1 fuel provides a good example, for which an extensive (and expensive) treatment program has been necessary to render the fuel eligible for disposal. This supports the case for the immediate treatment of fuel that has not yet degraded, as is the case for the HBWR fuel stored in Halden. Long term storage would require extensive resources, as it would be necessary to provide continuous surveillance, to maintain security and to plan for re-packing and perhaps construction of replacement storage facilities.

To maintain the technical assurances of safety and security of long term storage, it is essential to provide secure financial resources and stability for the organisations and agencies that are charged with carrying out and supervising the storage. An example of how changes in the political landscape can adversely affect the safety of spent fuel is the case of the Vinca fuel in Serbia (discussed in Chapter 6.8). In this case, the wars of the early 1990s impeded the ability of the relevant organisations to maintain the necessary storage conditions, with the result that the fuel is now severely degraded.

Another consideration related to storage and postponed decision is that, should new solutions not be developed, current solutions may no longer be available; for example, a reprocessing plant may shut down during the period of long-term storage and not be replaced. Additionally, while there currently exists in Norway extensive competence with regard to nuclear fuels, it is not known whether this would be the case after a storage period lasting for several decades.

The storage and postponed decision or “wait and see” option does not constitute a final disposal treatment, as it does not produce a waste form suitable for disposal, and for this reason was rejected by the Swedish authorities in the case of the R1 reactor fuel. Additionally, the Norwegian fuel has cooled to such an extent that it can no longer be considered as self-protecting (that is, would not deter those with malicious intent). Alternatives including long term storage do not satisfy proliferation requirements. The committee considers that this option does not meet the requirement that the solution should be in line with the ethical recommendations made by the OECD NEA.

8.1.3 Fuel exchange

The committee has considered the option of exchanging the metallic uranium and Al-clad fuels with different fuels from another country. It is of the opinion that the ethical considerations of transferring the responsibility to manage the fuel to another jurisdiction would be in many ways similar to those discussed above with respect to long term interim storage with postponed decision. Furthermore, it is not aware of any country that would benefit from accepting such fuels. Hence, it has taken no further steps to investigate the possibilities of specific fuel exchanges.

The committee considers that the fuel exchange option does not meet the requirement that the solution should be in line with the ethical recommendations made by the OECD NEA.

8.1.4 Return to the country of origin

The committee has contacted the United States Department of Energy (DOE) to investigate the possibility of shipping the HBWR and JEEP fuels to the USA under the current take back programs. After contacts made at the IAEA meeting on research reactor fuel storage in Dounreay, a letter was sent to the Director of Materials Science and Technology at the DOE's Savannah River Site (Appendix 3).

A reply was received from the Gap Material Program Manager of the National Nuclear Security Administration (Appendix 3). This letter stated that, although the DOE/National Nuclear Security Administration's Global Threat Reduction Initiative (GRTI) does have programs (i.e. the Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) and Gap programs) to receive foreign nuclear materials for disposition, they have very strict eligibility criteria. The vast majority of the Norwegian fuel does not meet these criteria and therefore cannot be dispositioned in the United States. The exception is the 60 JEEP I seed fuel rods (Chapter 3), which were produced from US supplied UO_2, which may be returnable under the FRRSNF program.

Notwithstanding the reply received from the DOE, the committee recommends that political contacts are made with the United States to further investigate this disposition option, given the close relations between the two countries and the strong support given by Norway to, for example, the Reduced Enriched Research and Test Reactor (RERTR) program and other non-proliferation activities.

8.1.5 Commercial processing using the PUREX technique

The PUREX process is the only spent fuel conditioning treatment that has been used on an industrial scale and that is offered as a commercial service. Reprocessing of civil nuclear fuel is performed in France, India, Japan, the Russian Federation and the UK. As described in Chapters 6.6 and 6.7, many countries have taken advantage of the programs that allow high enriched fuel to be returned to country of origin, i.e. the USA or the Russian Federation. Although the main objective of these two programs is to eliminate highly enriched uranium, participation implies the willingness of the consigning country to allow their fuel to be reprocessed. The U. S. Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Program and the Russian Research

Reactor Fuel Return (RRRFR) programs have resulted in the shipment of fuel from Argentina, Australia, Austria, Brazil, Canada, Chile, Columbia, Denmark, Finland, Germany, Greece, Indonesia, Italy, Japan, Netherlands, Phillipines, Portugal, Spain, Sweden, Switzerland, Taiwan, Thailand, Uruguay, Venezuela, Bulgaria, the Czech Republic, Kazakhstan, Latvia, Libya, Poland, Romania, Serbia, Uzbekistan and Vietnam. In the USA itself, for political reasons, extensive resources have been used over a thirty-year period to find an alternative to PUREX for conditioning of aluminium-clad spent fuels. However, it has now been concluded after all that PUREX is the most appropriate treatment.

Although PUREX is normally employed to reprocess fuel, there is no requirement to re-use any of the products, and hence the process can legitimately be designated as conditioning or processing according to the definitions given in Chapter 1.

Commercial (re)processing services are currently offered in France or Russia; it is not known how long this option will be available. The committee has contacted AREVA NC and Sosny Research Company to investigate the possibilities for processing the fuel in, respectively, France and Russia.

The chairman and secretary of the committee met with the Senior General Manager of the Recycling Business Unit of AREVA NC in France to discuss the feasibility of conditioning the metallic and Al-clad fuels. At this introductory meeting, information was given that there were no known reasons why AREVA could not treat the fuel, although there were many technical details to be clarified.

The main outcome of the meeting was that the committee’s request was to be presented to AREVA NC senior management to obtain a go/no go decision to continue negotiations. Ultimately, any treatment would require a political agreement between the French and Norwegian governments. Although it was understood on both sides that it was still too early to discuss such an agreement, the committee was informed that, in order to continue with its request, AREVA NC senior management required a signal from the Norwegian authorities to show their serious intentions in the matter. It was suggested that Norwegian politicians visit the French facilities in La Hague (the minutes of the meeting are attached in Appendix 4).

Following an approach from the chairman of the Technical Committee, the Ministry of Trade and Industry responded that such a visit could not take place at this stage as it could be construed as supporting the reprocessing option before a decision in principle has been taken on the preferred technical solution. Consequently, AREVA NC has informed the chairman of the committee that, under the present circumstances, AREVA NC will not pursue the possibility of processing the fuel in France.

The committee commissioned Sosny Research and Development Company to prepare a comprehensive report on the possibilities for handling SNF from Norwegian research reactors in the Russian Federation; the report is included as Appendix 5.

In summary, Russian laws allow import of Norwegian spent fuel to the Russian Federation for reprocessing and subsequent handling of the reprocessing products, and it is technically possible to process the fuels in the Mayak facility. As the first step in the preparation of an agreement, the authorized Norwegian organization must send to Rosatom State Corporation a letter of request to consider the possibility of importing the spent fuel to the Russian Federation for reprocessing, and stating whether the reprocessing products should be returned to Norway or disposed of in Russia.

The report describes in detail the approvals and authorisations required in the Russian Federation, together with the documents required to import SNF into Russia from Norway. Following Sosny visits to Halden and Kjeller, details are given of the required preparatory work to cut and repackage the fuel and of the facilities required to load the spent fuel into transport casks. Two options are given, using different casks, for all of which there exists experience in transporting SNF. Depending on the choice of cask, the fuel could be shipped in either one or three transports. The two options are summarised in the table below.

Option	**1**	**2**
Casks	SKODA VPVR/M and TUK-19	TUK-108/1
Number of transports	3	1
Repackaging	yes	no
Fuel assembly dismantling	yes	no
Fuel rod cutting	yes	no
New baskets for casks	no	yes
Massive equipment for cask transportation	no	yes
Loading in HBWR and JEEP II SNF storage buildings	yes	No
Time required, repackaging / loading	3 years / 8 months	Not required / 2 months
Earliest transport date	2012	2015

The main products of reprocessing the SNF would be:

Product	**Estimated mass**	**Estimated volume**
Uranyl nitrate hexahydrate ($UO_2(NO_3)_2{\cdot}6H_2O$)	23 tonnes	8.2 m^3
Plutonium dioxide (PuO_2)	1 – 1.5 kg	
Vitrified high-level radioactive waste	50 kg	20 litres

There are several options for final disposition of the reprocessing products, with the exception of plutonium, which under Russian law cannot be exported. One option is that all the products can be either used, or subjected to a final disposal, in Russia. Alternatively, the uranyl nitrate and/or the vitrified high level waste can be returned to Norway after interim storage in Russia. The duration of interim storage would be specified in the contract; preliminary estimates are six years for uranyl nitrate and 20 years for vitrified HLW. Usually, uranyl nitrate is transported by rail in special container railcars TK-VG-24 (20 TUK-24 casks per railcar). Canisters with vitrified radioactive waste can be transported in TUK-108/1 casks.

The costs of the above operations would be in the range from 38 to 58 million US dollars, depending on the shipment option and whether the processing products are returned to Norway. The cheapest option is transportation of the SNF in TUK-108/1 casks and return of regenerated uranium in the form of uranyl nitrate salt to Norway.

It should be noted that these costs do not include those for infrastructure upgrading at IFE Kjeller, or those for fuel handling operations (unloading, dismantling of the fuel assemblies, inspection, cutting, repacking and etc). A preliminary estimate of these costs is approximately 15 MNOK (3 million US dollars).

The inclusion of the Special Ecological Program in any contract with the Russian Federation should be emphasised. This is a program for rehabilitation of contaminated areas around Mayak that is supported by funds received from foreign trade operations with spent fuel; thus, new operations at Mayak are used to fund clean-up of contamination produced by previous operations. Details of specific projects are given in [Kastina and Golubkin, 2009]. The costs for the Special Ecological Program comprise 30 per cent of the total costs, and would thus be in the range from 11 to 17 million US dollars.

The committee is aware that a recommendation to process using the Russian facilities in Mayak may be controversial for many stakeholders because of the known environmental and safety problems associated with these facilities in the past [Bellona, 2004; Kastina and Golubkin, 2009].

In May 2009, a workshop was held by the IAEA Contact Expert Group on management of spent nuclear fuel and radioactive wastes [IAEA, 2009e]. The workshop was sponsored by the Norwegian Radiation Protection Authority (*Statens strålevern*). One objective was to share experiences and knowledge, in regard to Russian nuclear regulatory issues, from, among others, the perspectives of international donors to projects for removal of SNF from former navy bases in the North-West and Far-East of Russia. Norway is one of the international donors to this program, along with (among others) France, Canada, UK and Sweden. All SNF that technologically can be reprocessed will be reprocessed at Mayak: thus, Norway is currently providing funds for the reprocessing of SNF in Mayak.

Another objective was to discuss knowledge as to both environmental and societal impacts of the Mayak facility operations. The following statement was recorded in the minutes of the meeting:

> *International donors expressed confidence that given the information presented in regard to PA Mayak operations they were confident that projects for removal of SNF from former navy bases in the North-West and Far-East being funded or planned by international donors would not have any significant environmental impact.*

The committee considers that processing by a commercial supplier meets all the requirements listed at the beginning of this chapter: it is supported by the IAEA (for example, the return of Serbian SNF to Russia); the technology is mature and well proven; the final waste form does not adversely impact the environment; the technique offers value for money; and is in line with the decisions taken by several other countries on similar fuel types.

8.1.6 Conditioning in Norway

The committee was mandated to concentrate on processes that can be performed in Norway. Such a domestic treatment would have the advantage that full control of the process would be retained. If domestic treatment of the fuel is preferred, there are several options:

- A technology based on the PUREX process used in current reprocessing plants.
- A process based on electrometallurgical treatment, such as is used on metallic fuels from the EBR-II reactor in the USA. The existing technical literature is sufficient to begin investigation of this method; however, the committee has not investigated whether technological aid could be obtained from the operators in the USA.
- Calcination.
- Development of one of more techniques that are currently at very early stages of technical maturity, for example those based on furnace melting or chlorine volatility.
- A novel technique developed during a domestic research and development project.

In short, it can be stated that no facilities currently exist that can be used to condition metallic uranium fuel or aluminium-clad fuel. Constructing and operating facilities in Norway is fully feasible. Such a facility would require phases of design, construction, testing, operation and decommissioning. Additionally, it would produce significant amounts of radioactive waste products.

Conditioning using a development of PUREX

As originally developed, spent fuel separation techniques were true reprocessing operations, with the goal of recovering plutonium for military purposes. The same techniques were subsequently developed for research- and commercial reactor fuels, to avoid the need to treat as waste useful components of spent fuel. For breeder reactors, reprocessing was a central part of the fuel cycle. “Reprocessing” has become an emotive term, associated with weapons and radioactive discharges to the environment. Even though safeguards can be – and are – put in place to ensure that recovered materials cannot be used for military purposes, and current radioactive discharges are much lower than those allowed earlier, there are still many stakeholders with objections to the process.

However, it is important to differentiate between reprocessing as a strategic use of spent fuels and PUREX as a chemical engineering process. Dissolution of materials in acid and wet chemical separation techniques are not exclusive to the nuclear industry; PUREX was developed from standard chemical engineering processes. Chemical processes are often carried out preferably in liquid systems as homogeneous phases are produced which enable better control than in the case of reactions including solid reactants. Further, wet methods require lower temperatures and pressures than gaseous systems, and again allow better control.

Thus, the committee considers that, should a domestic treatment be preferred, a process based on PUREX should be chosen, as the general benefits of wet processes and the specific large experience base of PUREX outweigh the objections that may be made based on the association of the process with military purposes and environmental discharges. The current PUREX process can be adapted (as is being done in the development of next generation reprocessing plants) to avoid the production of a pure plutonium product stream, and discharges can be controlled. Thus, the fuel would be conditioned rather than reprocessed, according to the definitions given in Chapter 1.

However, if the choice is to be made between a domestic treatment and the purchase of commercial services, then the difference in price must be taken into account. It is considered that the costs of treating the fuel in a domestic facility would be at least fifty times higher than those of purchasing an existing, commercial treatment. In economical terms, the costs of constructing a domestic facility are unjustifiable if an acceptable commercial service is available.

In addition to the higher costs, the extra radioactive wastes that would be produced as a result of the decommissioning of the facility must be considered.

The committee consider that, if an acceptable commercial service is available, a domestic PUREX-type treatment option does not meet the requirements that it should offer value for money or that it should be environmentally sound. However, if a domestic treatment is necessary, a PUREX-type treatment is the recommended option.

Electrometallurgical conditioning (pyroprocessing)

The only known alternative to PUREX as a method to stabilise spent fuel, and which has been developed to the stage where it is used regularly, is the electrometallurgical treatment that has been developed in the USA specifically for EBR-II fuel. This fuel contains sodium, and is thus unsuitable for conditioning using PUREX. Some stakeholders may prefer this option as the process has the benefit of not producing a pure plutonium product. Others may prefer such a process for the reason that it is not “reprocessing” as the term is generally understood. However, this treatment is not offered on a commercial basis, and, as mentioned in Chapter 6.2, the technique could not be used on the Norwegian SNF without a pre-treatment step to remove the aluminum cladding. In purely technical terms, this process offers no advantages over a wet chemical technique for HBWR and JEEP fuels.

The committee considers that electrometallurgical treatment does not meet the requirements that the fuel management strategy should be technically suitable and offer value for money.

Calcination

As reported in Chapter 6.2, one suggestion – which was rejected – for management of SNF from the Hanford N reactor fuel was calcining; that is, dry oxidation of metallic uranium to UO_2. Some stakeholders may prefer this option as the process has the benefit of not producing a pure plutonium product. Others may prefer such a process for the reason that it is not “reprocessing” as the term is generally understood. However, this treatment is not offered on a commercial basis.

Some idea of the practicalities of calcination can be obtained from a patent application for a process for converting uranium metal alloys to UO_2 powder and pellets [EPO, 2000]. This process was designed for manufacture of nuclear fuel, and thus was to be applied to unirradiated uranium alloys. The starting material for the process was uranium alloy disks of one inch (2.5 cm) diameter and ¼ to ½ inches (0.6 to 1.2 cm) thick. (This would mean that, to apply the process to HBWR or JEEP fuels, the spent fuel rods would need to be de-clad and cut, requiring a large number of cuts per rod.) The disks would then be reduced in size using a jaw crusher to produce particles in the desired size range, such that they will be completely oxidised in a furnace (at 350 to 500 °C) to UO_2 powder.

The application noted that uranium metal is pyrophoric in air and oxygen. Spontaneous ignition occurs at a temperature of about 240 °C, depending on the rate at which the uranium surface oxidises in air balanced against the rate at which the heat of reaction is lost to the surroundings. Therefore, the particle size or the particle surface area per unit weight is an important part of the process. For the alloys considered in the application (uranium with Fe, Ni or Cr up to 15 weight percent), ignition did not occur at temperatures below 600 °C, but sparks, smoke and heat were observed during the size reduction operations. Since metallic uranium ignites at lower temperatures (240 °C), the risk of ignition of spent metallic uranium fuel would be higher than for the specified uranium alloys, and careful process control would be required.

The size of the process equipment is limited by criticality and radiological considerations (even in the case of unirradiated materials). For enrichments of 5 per cent, safe geometry limits were a cylindrical tank with diameters less than 10 inches (25 cm). Of course, for natural uranium, these limits would be higher. Radiological considerations impact the equipment design because operations must provide adequate containment of airborne radioactive materials.

The suitability of the calcining process is also affected by physical form of the end product - UO_2 powder. A powder has a much larger surface area than a high density material such as a vitrified waste product. Should groundwater come into contact with a powder in a repository, the leaching rate of fission products and actinides would be correspondingly higher. Such a process would probably require a further step to sinter the powder product to form a high density final product.

In purely technical terms, calcination offers no advantages over a wet chemical technique for HBWR and JEEP fuels. The committee considers that calcination does not meet the requirements that the fuel management strategy should offer value for money, and be technically mature and environmentally sound.

Table 8-1. Comparison of options for conditioning of metallic uranium and aluminium-clad spent fuels

Method	**IAEA and OECD NEA recommendations / ethical aspects**	**Technical suitability**	**Technology maturity / availability**	**International experience**	**Environmental impact**	**Value for money**
Direct disposal	Not applicable	Low: possible formation of pyrophoric and gaseous reaction products with water	No experience with type of fuels under consideration. Concept accepted in, among others, Sweden and Finland for ***oxide*** fuel.	None	Poor: possible formation of pyrophoric and gaseous reaction products with water. Possible release of radioactivity to the biosphere.	Good
Storage and postponed decision	Ethical objections	Low: degradation of fuel may occur	Mature: current situation	Extensive	Poor: degradation of fuel will occur	Poor: is not a final solution
Exchange of fuel	Ethical objections	Unknown: no potential exchange identified	Unknown: no potential exchange identified	Cases exist.	Unknown: no potential exchange identified	Unknown: no potential exchange identified
Return to the country of origin	Supported: eg US and Russian Return Programs	Mature and well-proven	Mature and well-proven	Mature and well-proven	Waste products stable in repository (fuel will be conditioned by PUREX).	Good
Commercial processing using PUREX	Supported: eg Serbian fuel, US and Russian Return Programs	Mature and well-proven	Mature and well-proven	Mature and well-proven	Waste products stable in repository (fuel will be conditioned).	Good: commercial services offered.

Method	**IAEA and OECD NEA recommendations / ethical aspects**	**Technical suitability**	**Technology maturity / availability**	**International experience**	**Environmental impact**	**Value for money**
Domestic conditioning using development of PUREX	Not applicable	PUREX is mature and well-proven. Developments underway internationally	PUREX is mature and well-proven. Developments underway internationally	PUREX is mature and well-proven. No experience with developments	Waste products stable in repository (fuel will be conditioned). Additional radioactive waste from domestic facility	Very poor: domestic facility required
Domestic conditioning using electrometallurgical conditioning	Not applicable	Medium: requires extra step for Al removal	Clad removal step unproven	Metallic U fuel with stainless steel cladding (EBR-II fuel)	Waste products probably stable in repository	Very poor: domestic facility required
Domestic conditioning using calcination	Not applicable	Low: requires decladding and extensive cutting of fuel rods, plus further conditioning to produce a stable waste form.	Not proven	Tested on unirradiated material	Waste products would be stable in repository if suitable conditioning method identified.	Very poor: domestic facility required

8.2 Storage

The decision on the type of storage facility should be taken based on the physical, chemical and radiological properties of waste form(s) that the facility will store. The committee assumes that standard HBWR fuel (UO_2 fuel clad in Zircaloy), which is not covered in the Technical Committee's mandate, will be stored in its current physical form, i.e. as spent fuel assemblies. However, if the metallic uranium and aluminium-clad fuels are conditioned as recommended in Chapter 9, the conditioning products will require storage if some or all of them are returned to Norway. Spent fuel storage must be safe against a criticality incident, while vitrified waste canisters may have higher heat generation and/or radiation levels. A facility to store both spent fuel and vitrified high level waste has been constructed in the Netherlands (Chapter 7.1).

Further, the committee assumes that a dry storage facility will be chosen. The committee recommends against an underground facility with its associated risks of water ingress. As described in Chapter 7.1, there are thus three options for the choice of a dry storage facility:

- Casks.
- Vaults.
- Silos.

The committee recommends that the choice of facility type should be made based on a full economical analysis of the three options. It also notes that there are several commercial organisations that offer such analyses.

9 RECOMMENDATIONS

Based on the analysis of the various options for the management of the Norwegian spent metallic uranium and aluminium-clad fuel presented in Chapter 8, the committee makes the following recommendations:

1. Metallic uranium is pyrophoric, and reacts with water to form hydrogen gas and pyrophoric uranium hydride. Aluminium cladding reacts with water to form hydrogen gas, and can undergo severe localised corrosion, which can penetrate the cladding and expose the metallic uranium fuel. Since metallic uranium fuel and/or fuel clad in aluminium are unstable, they should be stabilised before interim storage and disposal. The method chosen to stabilise the fuels should render them eligible for both interim storage and consequent disposal: i.e. such that further treatment of the fuel after interim storage is not required.

2. The storage and postponed decision or “wait and see” option does not constitute an endpoint for the fuel. The committee considers that this option represents a major uncertainty/deficiency since it presumes stability of future societies and their continuing ability to carry out the required safety and institutional measures, and since Norway is not working actively to investigate or develop alternative solutions, such as solutions based on technology development or multi-national facilities. In the light of these objections, and in-line with the Swedish decision on management of the R1 reactor fuel, the committee recommends against this option for the Norwegian spent fuel.

3. Similar objections apply to consideration of exchanging fuel with another country, as the responsibility for managing the fuel is transferred. The committee thus recommends against this option for the Norwegian spent fuel.

4. The small volume of spent nuclear fuel in Norway makes it difficult to justify the costs of constructing a domestic conditioning facility as long as overseas commercial services are available that satisfy financial, political, and environmental requirements.

5. The committee recommends that the Norwegian government immediately makes a high-level contact with the United States government to discuss the possibility of shipping the fuel to the United States under the current take back programs since these will cease in 2016.

6. The committee recommends that the Norwegian government sends to Rosatom State Corporation a letter of request to consider the possibility of importing the spent fuel to the Russian Federation for processing. Following the 2009 report of the IAEA Contact Expert Group on management of spent nuclear fuel and radioactive wastes, and in view of the fact that the Norwegian government is donating funds which contribute to processing of Russian fuel in Mayak, the committee considers that this option satisfies environmental and political requirements. The committee recommends that, if a decision is taken to condition the fuel using Russian commercial services, it should be taken as early as possible in order to take advantage of the current availability of these services. This availability is linked to the activities of the present Russian take back program.

7. The committee recommends that the Norwegian government take the necessary steps to acquire a report from AREVA NC in France summarising the feasibility of processing the spent fuel in the Cap La Hague facilities. The report should include technical feasibility, cost and options for management of the processing products.

8. The decision on the method of interim storage of the conditioning products following treatment of the metallic fuel should be taken after the decision on the stabilisation method. The choice of storage method should be based on the physical, chemical and radiological properties of the waste form(s) together with an economical analysis of the three options. The committee recommends dry storage in casks, silos or vaults.

10 CONCLUDING REMARKS AND NEXT STEPS

Inspection of the committee's mandate shows that it was envisaged that there would be a range of available treatment options, which could be compared with each other and for which the volumes and types of material to be stored could be calculated. However, in practice, once recommendations were made against direct disposal and interim storage with postponed decision, one treatment option – overseas commercial processing – is clearly shown to be superior when compared against the chosen selection criteria.

Thus, the next step should be a high level political decision on whether to pursue the overseas commercial processing option. If this option is to be progressed, a decision must also be taken on whether to take back the conditioning products where there is an option to dispose of them in the processing country. For processing at Mayak, there are options to store and dispose of all waste products in Russia, whilst French law prescribes return of high level waste. Some of the countries participating in the Russian Research Reactor Fuel Return Program and all of the countries participating in the U. S. Foreign Research Reactor Spent Nuclear Fuel Acceptance Program will store and dispose of all waste products in Russia and the U. S., respectively.

If, instead, a domestic conditioning treatment is chosen, a decision on the method will be required (for example, wet processing using PUREX or calcination). In all cases, approximately 20 tons (10 m^3) of uranium product and 1.5 kg of plutonium would be produced, but the high level, secondary and decommissioning wastes would be dependent on the method.

Only when the waste forms and volumes are known can recommendations be made for storage methods; and, as discussed above, it is necessary first to take a fundamental decision regarding the conditioning method.

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APPENDIX 1. MANDAT FOR TEKNISK UTVALG

12.12.07

Mandat for teknisk utvalg

Utredning av spesialbehandling av ustabilt brukt brensel
- Lagring og deponering av metallisk uran og brensel med aluminiumskapsling

Teknisk utvalg skal gi en faglig utredning av spesialbehandling av ustabilt brukt brensel, innenfor samfunnsmessige, samfunnsøkonomiske og miljømessig akseptable rammer. Teknisk utvalg skal legge vekt på løsninger som kan gjennomføres i Norge. Ved utredning av alternativer som innebærer utenlandske løsninger skal utvalget legge til grunn regjeringens målsetning om å få lagt ned anlegget ved Sellafield. Utvalget bør unnlate å utrede løsninger som åpenbart ikke vil kunne hente tilstrekkelig støtte i gjennomføringsfasen. Det er ønskelig at utvalget sammenligner kostnadene ved spesialbehandling av brukt brensel i Norge med kostnadene ved spesialbehandling av norsk brukt brensel i utlandet.

1. **Bakgrunnsinformasjon**
    - Identifisere relevante problemstillinger mht. lagring og deponering, av metallisk uran og brensel med aluminiumskapsling. Det skal særlig rettes fokus på de nevnte materialers stabilitet ved lagring og deponering. Det bør gis en kort oppsummering av faktagrunnlaget fra fase 1-utvalgets rapport knyttet til metallisk uran og brensel med aluminiumskapsling.

2. **Problemstillinger knyttet til lagring og deponering av metallisk uran og brensel med aluminiumskapsling**
    - Utvalget bør gi en grundig beskrivelse av egenskapene til metallisk uran, med særlig fokus på stabilitet og mulige endringer i egenskapene ved lagring og deponering.

    - Utvalget bør gi en grundig beskrivelse av egenskapene til aluminiumskapsling, med særlig fokus på stabilitet og mulige endringer i egenskapene ved lagring og deponering.

3. **Internasjonale anbefalinger og trender**
    - Hvilke anbefalinger/retningslinjer gir IAEA mht. lagring og deponering av metallisk uran og brensel med aluminiumskapsling?

    - Hvilke nasjoner har erfaring med metallisk uran og hva er (planlagt) gjort relatert til lagring og eventuell deponering av metallisk uran?

    - Samme som i punktet over, men for brensel med aluminiumskapsling.

4. **Lagring og deponering av metallisk uran**
    - Identifisere metoder for sikker lagring og deponering av metallisk uran og antyde kostnader og avfallsmengder forbundet med de identifiserte metodene.

    - Vurdere alternativer til lagring og deponering og antyde kostnader og avfallsmengder forbundet med alternative metoder.

5. **Lagring og deponering av brensel med aluminiumskapsling**

- Identifisere metoder for sikker lagring og deponering av brensel med aluminiumskapsling og antyde kostnader og avfallsmengder forbundet med de identifiserte metodene.

- Vurdere alternativer til lagring og deponering antyde kostnader og avfallsmengder forbundet med alternative metoder.

6. Status for metallisk uran og brensel med aluminiumskapsling.

- Utvalget bør identifisere mulige returordninger for brukt brensel med utgangspunkt i opprinnelsesland for det metalliske uranet som er benyttet i de norske forsøksreaktorene.

- Utvalget bør beskrive det metalliske uranets bestrålingshistorikk, med bakgrunn i informasjon fra IFE og Statens strålevern.

- Utvalget bør beskrive bruk av aluminiumskapsling i forskningsreaktorene i Norge, med bakgrunn i informasjon fra IFE og Statens strålevern.

- Utvalget bør lage en oversikt over enheter, vekt, volum lagret i Norge, med bakgrunn i fase 1-utvalgets rapport samt informasjon fra IFE og Statens strålevern. Herunder bør utvalget også estimere årlig tilvekst av brensel med aluminiumskapsling.

7. Leveranse

- Teknisk utvalg skal levere en rapport senest ett år etter at utvalget er blitt nedsatt. Rapporten skal være innspill og underlag for fase 2-utvalget. Det bør lages en norsk oversettelse av sammendraget til teknisk utvalgs rapport.

8. Oppsummering

Utvalget skal:

- Sammenfatte kostnader for de ulike behandlingsalternativene.

- Sammenfatte hvilke krav de anbefalte behandlingsalternativene stiller til mellomlageret og framtidig deponi.

- Anbefale behandlingsalternativer av ustabilt brukt brensel.

APPENDIX 2. MANDATE OF THE TECHNICAL COMMITTEE

Mandate for Technical Committee

Specification of treatment of unstable spent fuel – storage and disposal of metallic uranium and fuel with aluminium cladding

The Technical Committee shall produce an informed evaluation of possible treatments of unstable spent fuel, within acceptable societal, economical and environmental boundaries. The committee shall concentrate on processes that can be performed in Norway. If specifying processes that would be conducted in foreign countries, the committee shall be mindful of the Norwegian Government's aim that the Sellafield complex be shut down. The committee should avoid proposing solutions that will obviously not obtain necessary and sufficient support. It is desirable that the committee compares the costs of treatment of spent fuel in Norway with those for treatment of Norwegian fuel in foreign countries.

1. **Background information**

- Identify possible methods of storage and disposal of metallic uranium and fuel with aluminium cladding. Particular attention should be given to the materials' stability under storage and disposal. A short summary should be given of the factual basis from the Phase 1 committee's report on metallic uranium and fuel with aluminium cladding.

2. **Issues related to storage and disposal of metallic uranium and fuel with aluminium cladding**

- The committee should produce a thorough description of the properties of metallic uranium, focussing on its stability and possible changes under storage and disposal.

- The committee should produce a thorough description of the properties of aluminium cladding, focussing on its stability and possible changes under storage and disposal.

3. **International recommendations and trends**

- What recommendations and/or guidelines are issued by the IAEA with regard to storage and disposal of metallic uranium and fuel with aluminium cladding?

- Which countries have experience with metallic uranium, and what plans have been made, or actions taken, related to storage and disposal of metallic uranium?

- The same as for the above point, but for fuel with aluminium cladding?

4. **Storage and disposal of metallic uranium**

- Identify methods for secure storage and disposal of metallic uranium, and estimate costs and waste volumes associated with the different methods.

- Evaluate alternative methods for storage and disposal, and estimate costs and waste volumes associated with the different methods.

5. **Storage and disposal of fuel with aluminium cladding**

- Identify methods for secure storage and disposal of fuel with aluminium cladding, and estimate costs and waste volumes associated with the different methods.

- Evaluate alternative methods for storage and disposal, and estimate costs and waste volumes associated with the different methods.

6. **Status of metallic uranium and fuel with aluminium cladding**

- The committee should identify possible arrangements for return to the country of origin of spent fuel that has been irradiated in Norwegian research reactors.

- The committee should describe the irradiation history of metallic uranium fuel in Norwegian research reactors, from information to be supplied by IFE and the Norwegian Radiation Protection Authority (*Statens strålevern*).

- The committee should summarise the use of aluminium cladding in Norwegian research reactors, from information to be supplied by IFE and the Norwegian Radiation Protection Authority.

- The committee should produce a summary of the number of fuel rods, mass, and volume stored in Norway, from information from the Phase 1 report and from information to be supplied by IFE and the Norwegian Radiation Protection Authority. This summary should include an estimation of the annual increase in the amount of fuel with aluminium cladding.

7. **Delivery**

- The committee shall deliver a report at the latest one year after the appointment of the committee. This report will be the basis for the work of the Phase 2 committee. The summary of the report should be translated into Norwegian.

8. **Summary**

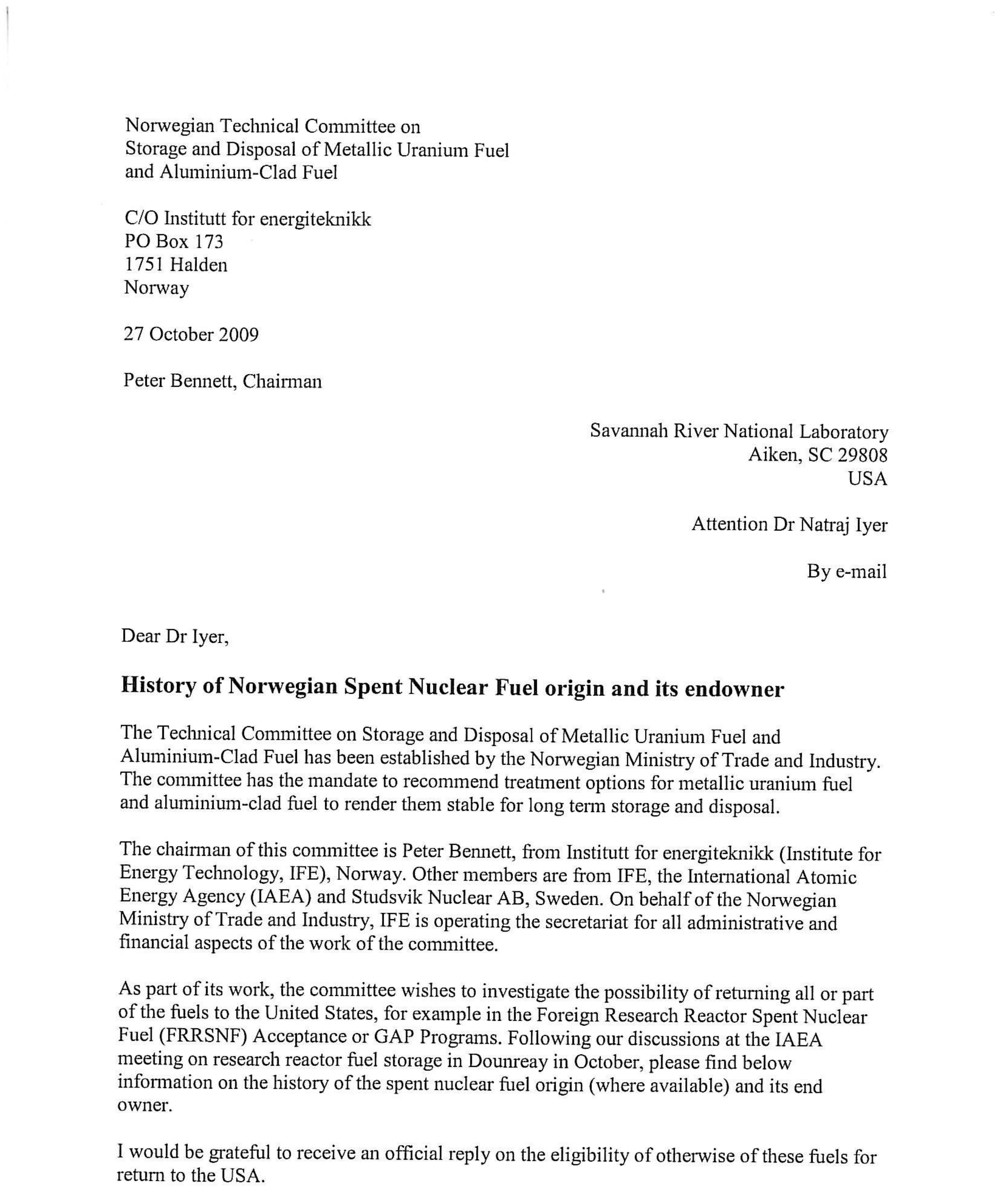
The technical committee shall:

- Summarise costs for the alternative treatment methods.

- Summarise the requirements of the alternative treatment methods with regard to intermediate storage and final disposal.

- Recommend treatment methods for unstable spent fuel.

APPENDIX 3. CORRESPONDENCE WITH THE UNITED STATES DEPARTMENT OF ENERGY



Norwegian Technical Committee on
Storage and Disposal of Metallic Uranium Fuel
and Aluminium-Clad Fuel

C/O Institutt for energiteknikk
PO Box 173
1751 Halden
Norway

27 October 2009

Peter Bennett, Chairman

Savannah River National Laboratory
Aiken, SC 29808
USA

Attention Dr Natraj Iyer

By e-mail

Dear Dr Iyer,

History of Norwegian Spent Nuclear Fuel origin and its endowner

The Technical Committee on Storage and Disposal of Metallic Uranium Fuel and Aluminium-Clad Fuel has been established by the Norwegian Ministry of Trade and Industry. The committee has the mandate to recommend treatment options for metallic uranium fuel and aluminium-clad fuel to render them stable for long term storage and disposal.

The chairman of this committee is Peter Bennett, from Institutt for energiteknikk (Institute for Energy Technology, IFE), Norway. Other members are from IFE, the International Atomic Energy Agency (IAEA) and Studsvik Nuclear AB, Sweden. On behalf of the Norwegian Ministry of Trade and Industry, IFE is operating the secretariat for all administrative and financial aspects of the work of the committee.

As part of its work, the committee wishes to investigate the possibility of returning all or part of the fuels to the United States, for example in the Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance or GAP Programs. Following our discussions at the IAEA meeting on research reactor fuel storage in Dounreay in October, please find below information on the history of the spent nuclear fuel origin (where available) and its end owner.

I would be grateful to receive an official reply on the eligibility of otherwise of these fuels for return to the USA.

Reactor Fuel type	**Fuel**	**Enrichment done / delivery of fuel powder**	**Producer**	**Current owner**
JEEP I				
Standard element / void element	Metallic natural U	Not applicable – not enriched	United Kingdom France Netherlands Norway	IFE, Norway
Seed elements	1.7% UO_2	USA	IFE, Norway	IFE, Norway
HBWR				
1st charge	Metallic natural U	Not applicable – not enriched	United Kingdom	IFE, Norway
JEEP II				
1st charge	3.5% UO_2		IFE, Norway	IFE, Norway
2nd charge	3.5% UO_2		IFE, Norway	IFE, Norway
3rd charge	3.5% UO_2		IFE, Norway	IFE, Norway
4th charge	3.5% UO_2		IFE, Norway	IFE, Norway
5th charge	3.5% UO_2		IFE, Norway	IFE, Norway
6th charge	3.5% UO_2	Germany	IFE, Norway	IFE, Norway
7th charge	3.5% UO_2	Sweden	IFE, Norway	IFE, Norway

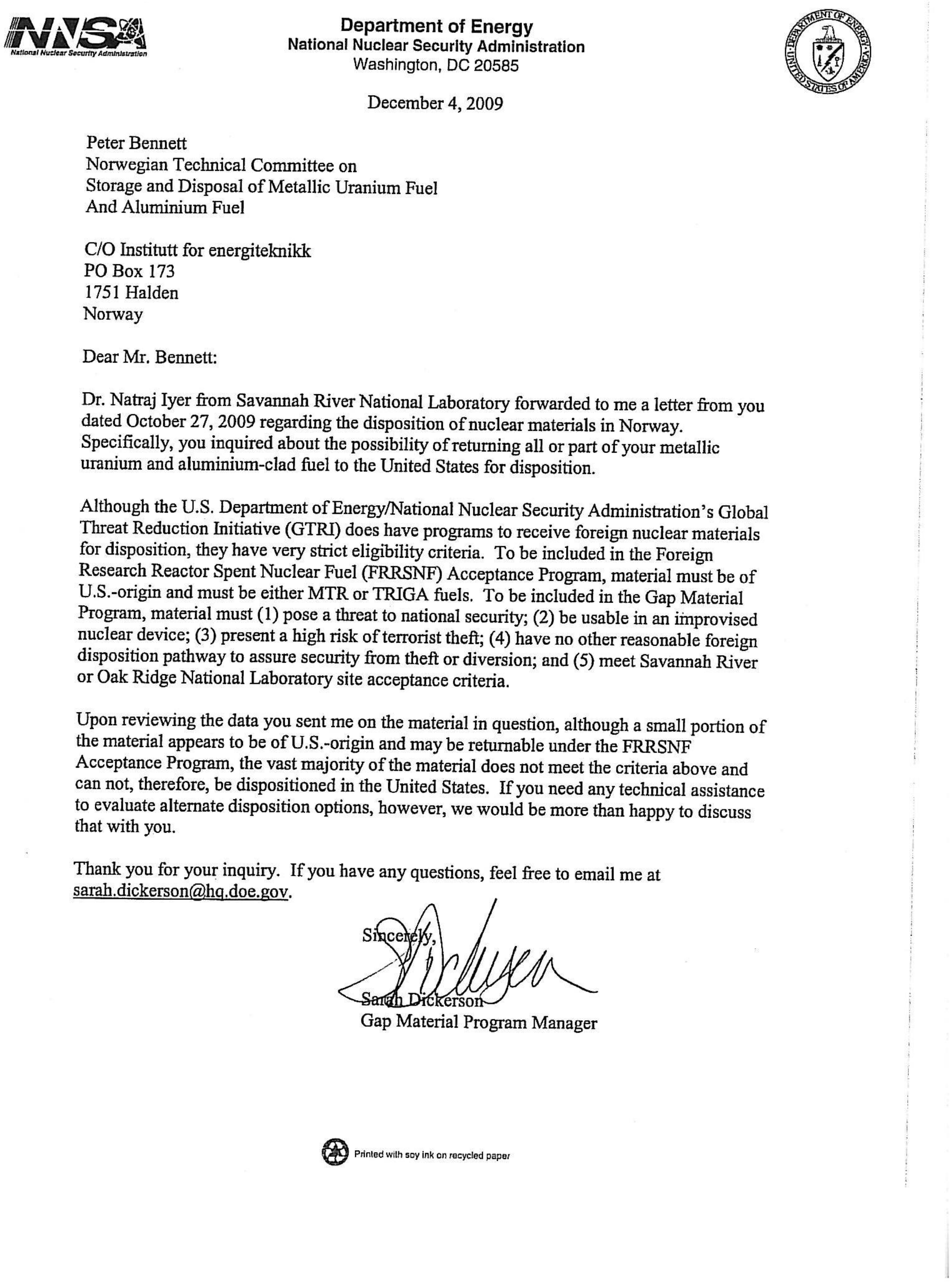
With best regards,

Peter Bennett
Telephone: +47 69212345
e-mail: peter.bennett@hrp.no

Copies

Mr Per Ivar Wethe
Secretary, Technical Committee on Storage and Disposal of Metallic Uranium Fuel and Aluminium-Clad Fuel

Dr Barbara Oberländer, Member, Technical Committee on Storage and Disposal of Metallic Uranium Fuel and Aluminium-Clad Fuel



Department of Energy
National Nuclear Security Administration
Washington, DC 20585

December 4, 2009

Peter Bennett
Norwegian Technical Committee on
Storage and Disposal of Metallic Uranium Fuel
And Aluminium Fuel

C/O Institutt for energiteknikk
PO Box 173
1751 Halden
Norway

Dear Mr. Bennett:

Dr. Natraj Iyer from Savannah River National Laboratory forwarded to me a letter from you dated October 27, 2009 regarding the disposition of nuclear materials in Norway. Specifically, you inquired about the possibility of returning all or part of your metallic uranium and aluminium-clad fuel to the United States for disposition.

Although the U.S. Department of Energy/National Nuclear Security Administration's Global Threat Reduction Initiative (GTRI) does have programs to receive foreign nuclear materials for disposition, they have very strict eligibility criteria. To be included in the Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Program, material must be of U.S.-origin and must be either MTR or TRIGA fuels. To be included in the Gap Material Program, material must (1) pose a threat to national security; (2) be usable in an improvised nuclear device; (3) present a high risk of terrorist theft; (4) have no other reasonable foreign disposition pathway to assure security from theft or diversion; and (5) meet Savannah River or Oak Ridge National Laboratory site acceptance criteria.

Upon reviewing the data you sent me on the material in question, although a small portion of the material appears to be of U.S.-origin and may be returnable under the FRRSNF Acceptance Program, the vast majority of the material does not meet the criteria above and can not, therefore, be dispositioned in the United States. If you need any technical assistance to evaluate alternate disposition options, however, we would be more than happy to discuss that with you.

Thank you for your inquiry. If you have any questions, feel free to email me at sarah.dickerson@hq.doe.gov.

Sincerely,

Sarah Dickerson
Gap Material Program Manager

APPENDIX 4. MINUTES OF MEETING HELD WITH AREVA NC

Technical Committee
Secretariat

Minutes from Meeting between AREVA NC and the Technical Committee on Storage and disposal of Metallic Uranium Fuel and Al-clad Fuels

Date: July 7, 2009

At present: Philippe Auziere, Recycling Business Unit New Projects - Senior General Manager

Peter Bennett, Chairman, Technical Committee on Storage and Disposal of Metallic Uranium Fuel and Al-clad Fuels

Per Ivar Wethe, Secretary, Technical Committee on Storage and Disposal of Metallic Uranium Fuel and Al-clad Fuels

Purpose of the Meeting:

Discuss the possibilities for reprocessing spent research reactor fuel –metallic natural uranium Al-clad fuel and JEEP II fuel (about 13 tons HM).
Background information on the Technical Committee and the types of fuels in question is given in the Appendix.

Outcome of the Meeting

- There were no obvious reasons why the fuels can not be reprocessed by AREVA NC.
- There are many technical details to be clarified. The most important are the supplier, manufacturing route and chemical composition of the fuel. If this information is not available then IFE must conduct dissolution trials of the metallic fuels in cooperation with CEA and AREVA NC. Visual inspection is required to confirm status of the used fuels.
- A political agreement between the French and Norwegian government will be required
- Transport would be arranged by AREVA NC through a subcontractor. This would most likely be by boat transport.

Separated Products

- The recommended option of AREVA NC is to incorporate the separated plutonium into MOX fuel for commercial power nuclear reactors. This plutonium will be under full IAEA/EURATOM safeguards. However, if required, arrangements can be made for return of plutonium to Norway.
- According to French law, the high level waste must be returned to Norway. The waste will be vitrified and returned in universal French containers.
- The separated uranium will be transferred to the ownership of AREVA NC.

Outline Schedule

- Contract signed (Jo): 2012 – 2015
- Shipment of spent fuel (J1): Jo + (3 - 6) years
- Interim storage (J2): J1 + (3 - 6 years)
- Chemical processing (J3): J2 + (3 - 6 years)
- Return of first batch of high level waste from around at the latest 2030.

Miscellaneous

- It is much easier to manage non-degraded fuel than fuel that has corroded. In the case that the used fuels are degraded, the safety analysis has to take into account the risk of the activity leakage and fissile material dissemination.
- The Technical Committee informed AREVA NC of its January 15, 2010 deadline. This was noted by AREVA NC however it may not be possible to obtain the required approval by this deadline.
- The Technical Committee agreed with the suggestion from AREVA NC that, to show the serious interest of the Norwegian side, the Technical Committee will strongly recommend to relevant Norwegian Ministries that a visit is arranged to AREVA NC fuel processing facilities in La Hague, France.

Next Steps

- M. Auziere will present our request to AREVA NC senior management to obtain a go/no go decision to continue negotiations.
- If a positive decision is taken by AREVA NC, the next steps will be specified by AREVA NC with CEA and the Technical Committee.

Appendix

OUTLINE INFORMATION TO ACCOMPANY THE NORWEGIAN REQUEST FOR DISCUSSIONS WITH AREVA CONCERNING THE POSSIBILITIES OF CHEMICAL PROCESSING OF SPENT RESEARCH REACTOR FUEL IN FRANCE

Prepared by:	Peter Bennett, Chairman, Technical Committee on Storage and Disposal of Metallic Uranium Fuel and Al-clad Fuels;
	Per Ivar Wethe, Secretary, Technical Committee on Storage and Disposal of Metallic Uranium Fuel and Al-clad Fuels

BACKGROUND

Two committees have been established to make recommendations for the interim storage and final disposal of spent fuel from three research reactors that have been operated in Norway, at the Kjeller and Halden sites:

- JEEP I, Kjeller, 1951 – 1967.
- JEEP II, Kjeller, 1966 – present.
- Halden Boiling Water Reactor (HBWR), Halden, 1959 – present.

All existing spent nuclear fuel (SNF) is currently stored at the two sites. Of this fuel, several tons consist of (i) metallic uranium fuel clad in aluminium and (ii) UO_2 fuel clad in aluminium. Since both metallic uranium and aluminium may react with water, forming unstable reaction products (for example uranium hydride) and hydrogen gas, these fuels should be stabilised before they can be stored and placed in a final repository.

A summary of the metallic uranium fuel and aluminium-clad fuel is appended.

In January 2009, the so-called Phase-2 committee was established by Royal decree with the mandate to find the most suitable technical solution and localisation for intermediate storage for spent nuclear fuel and long-lived waste. The committee should evaluate different solutions and their associated costs, and then recommend one of the options. An important point, of crucial importance for the current discussions, is that, if possible, the need to repack the fuel before final disposal should be avoided.

The committee will base its work on the work of previous committees and on the report of the Technical Committee (see next paragraph).

Co-incident with the formation of the Phase 2 Committee, the Technical Committee on Storage and Disposal of Metallic Uranium Fuel and Al-clad Fuels was formed. This committee has the mandate to recommend treatment options for metallic uranium fuel and aluminium-clad fuel to render them stable for long term storage and disposal.

The chairman of this committee is Peter Bennett, from Institutt for energiteknikk (Institute for Energy Technology, IFE), Norway. Other members are from IFE, the International Atomic Energy Agency (IAEA) and Studsvik Nuclear AB, Sweden. On behalf of the Norwegian Ministry of Trade and Industry, IFE is operating the secretariat for all administrative and financial aspects of the work of the Technical and Phase-2 committees.

The recommendations of this technical committee will be delivered to the Phase 2 committee, which will recommend the technical option and location of an interim storage facility. However, since it is preferred that no further treatment of the waste is required when transferring it from interim storage to final disposal, in practice this means that the fuel should be stabilised in such a way that it is acceptable for final disposal.

SPECIFICATION OF SPENT NUCLEAR FUEL FROM NORWEGIAN RESEARCH REACTORS: METALLIC URANIUM FUEL AND ALUMINIUM-CLAD FUEL

1. HBWR FUEL

Fuel material	Metallic uranium
Fuel rod diameter, cm	2.54
Fuel length, cm	237.5
Number of fuel rods	308
Initial enrichment	Natural uranium
Burn-up, MWd/t U	12
Total mass U, tons	6.7
Cladding material	Aluminium
Irradiation period	1000 hours (1959 – 1962)
Storage	Horizontal dry storage, inside stainless steel capsules at IFE, Halden site

2.1 JEEP I FUEL

Fuel type	Metallic uranium
Fuel rod diameter, cm	2.54 cm
Fuel length, cm	190 cm (active fuel stack)
Number of fuel rods	460
Initial enrichment	Natural uranium
Burn-up, MWd/t U	1 – 1000 (mostly 200 – 400)
Total mass U, tons	3.1
Cladding material	Aluminium
Irradiation period	1951 - 1967
Storage	Vertical dry storage, inside stainless steel capsules at IFE, Kjeller site

2.2 JEEP I FUEL – SEED Fuel

Fuel material	UO_2
Fuel rod diameter, cm	1.27 cm
Fuel length, cm	96.5 cm (active fuel stack)
Number of fuel rods	60
Initial enrichment, % U-235	1.7% enriched UO_2
Burn-up, MWd/t U	570 -1000
Total mass U, kg	Approx 75
Cladding material	Aluminium
Irradiation period	1951 – 1967
Storage	Dry storage at IFE, Kjeller site

2.3 JEEP I FUEL – VOID elements

Fuel material	Metallic Uranium
Fuel rod diameter, cm	2.54
Fuel length, cm	190 cm
Number of fuel rods	7
Initial enrichment, % U-235	Natural
Burn-up, MWd/t U	unknown
Total mass U, kg	37
Cladding material	Aluminium
Irradiation period	1951 - 1967
Storage	Dry storage at IFE, Kjeller site

3. JEEP II FUEL

Fuel material	UO_2
Fuel rod diameter, cm	1,28 cm
Fuel length, cm	90
Number of fuel rods	1050
Initial enrichment, % U-235	3.5
Burn-up, MWd/t U	200-450
Total mass U, tons	Approx 1.0
Cladding material	Aluminium
Irradiation period	1966 to present
Storage	Dry storage at IFE, Kjeller site