PRESENTATION OF CEA NUCLEAR SERVICE FACILITIES FOR SPENT FUELS AND NUCLEAR WASTES

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ABSTRACT

The Nuclear Energy Division of CEA has several operational hot laboratories and experimental reactors, as well as old facilities under decommissioning. All this creates solid and liquid radioactive wastes and spent fuels which need corresponding service facilities for treating them. This paper gives an overview of the main service facilities. Concerning spent fuels, the main route is reprocessing to value their energetic content. However some specific spent fuels are put inside interim storage facility in Cadarache. Liquid wastes are treated by specific installations in each nuclear centre (Marcoule, Saclay and Cadarache). These buildings are under refurbishment and the cementation process is due to replace the bitumen used to embed waste sludge. Solid wastes are either send to ANDRA low level waste repository or put into interim storage facilities located on CEA sites awaiting the opening of ANDRA underground final repositories. In any case, optimisation of waste treatment remains a constant preoccupation for facility managers.

1 Introduction

The CEA Nuclear Energy Division gathers nearly all nuclear facilities for civil applications within CEA. Our nuclear facilities includes experimental reactors, such as the Osiris Material Testing Reactor in Saclay (to be replaced by RJH under construction in Cadarache), several hot laboratories: LECA-STAR, LEFCA, CHICADE in Cadarache, ATALANTE in Marcoule and LECI in Saclay [1]. We are also responsible for the decommissioning of some nuclear sites, like Fontenay-aux-Roses near Paris, Grenoble and the UP1 reprocessing plant in Marcoule (Fig 1).

These R&D nuclear facilities under operation and the ones under decommissioning are producing radioactive liquid and solid wastes, as well as spent fuels.

To deal with these wastes and fuels, CEA possesses a fleet of facilities for nuclear services (Fig 2). But most of them were built at the end of the fifties or in the sixties, and they need to be adapted to new safety standards, to waste production evolutions and to waste characterizations and process evolutions. The evolution of these facilities is also depending on the opening date of the underground ANDRA national repository (estimated 2025) and the corresponding specifications for the final form of waste packages. So basically, most service facilities dealing with wastes should be either refurbished or changed by new ones.

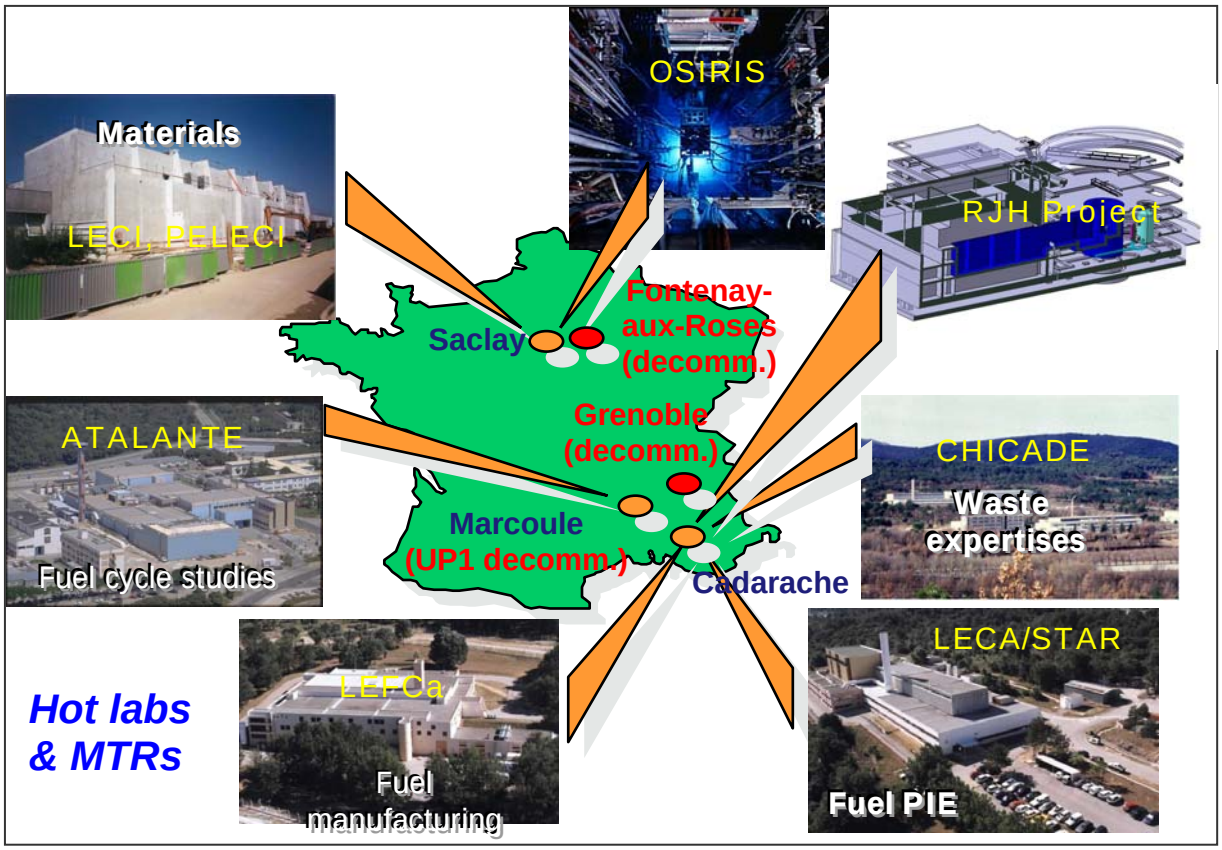

Fig 1. CEA hot laboratories, MTR and sites under decommissioning

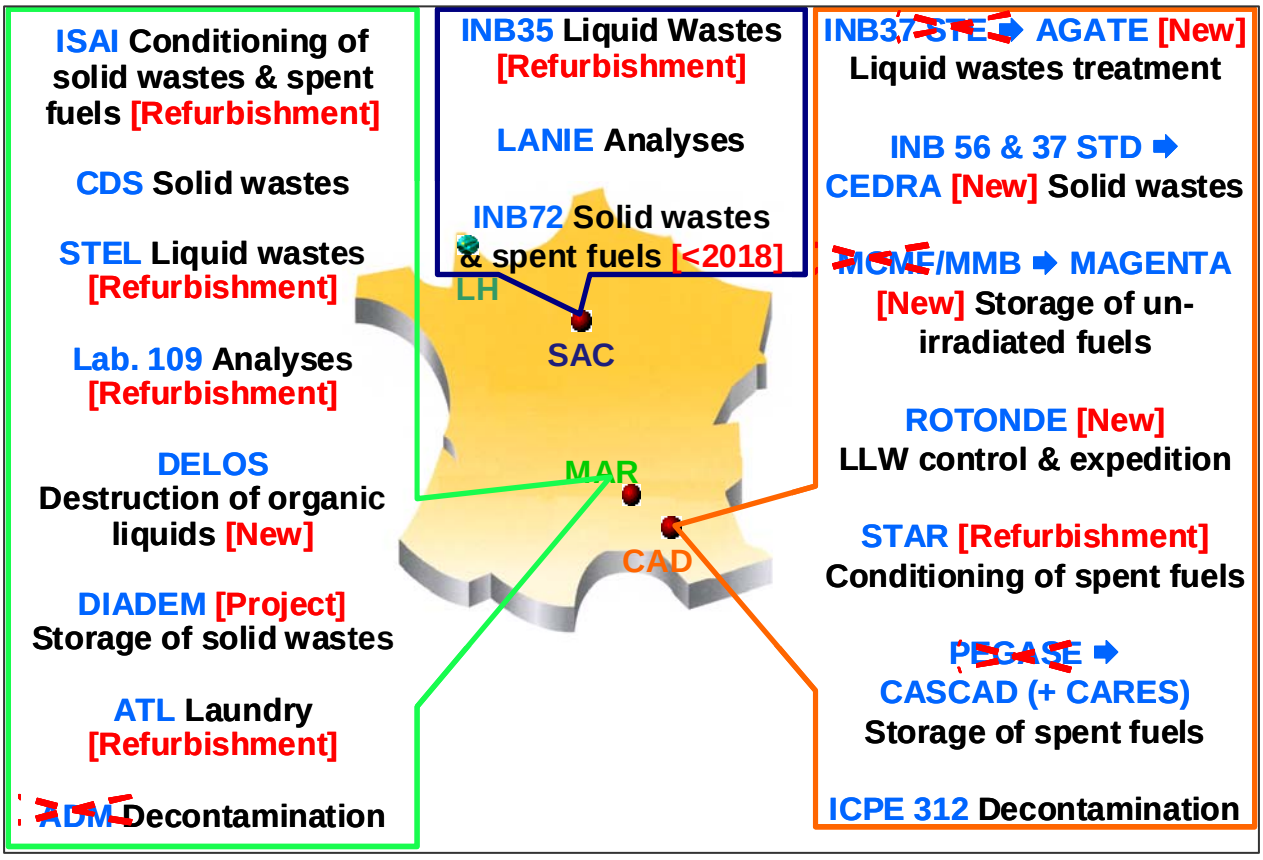

Fig 2. Facilities for nuclear services: waste treatment, decontamination, interim storages

2 Service facilities for spent fuels & un-irradiated fuels

2.1 Spent fuels

The general strategy concerning spent fuels consists to send them for reprocessing in the La Hague AREVA plant. This is typically the case for most fuels retrieved from Osiris MTR and will be also the case for fast breeder pins from Phénix. Spent fuels are not considered as wastes, but as possibly valuable energetic materials.

However, when considering small quantities of spent fuels, for instance those embedded in epoxy for metallographic examination after irradiation or for exotic spent fuels the processing of which is not demonstrated, it is not economically or technically reasonable to treat them in La Hague.

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CEA has two facilities for conditioning fuels inside canisters:

- STAR in Cadarache, which is mainly devoted to spent fuels from NPP, either old ones coming from graphite gas reactors from the French 1st generation of NPP, or spent fuels coming from R&D surveillance programmes for PWR of the EDF nuclear fleet.
- ISAI in Marcoule, which is mainly devoted to spent fuels from Phénix.

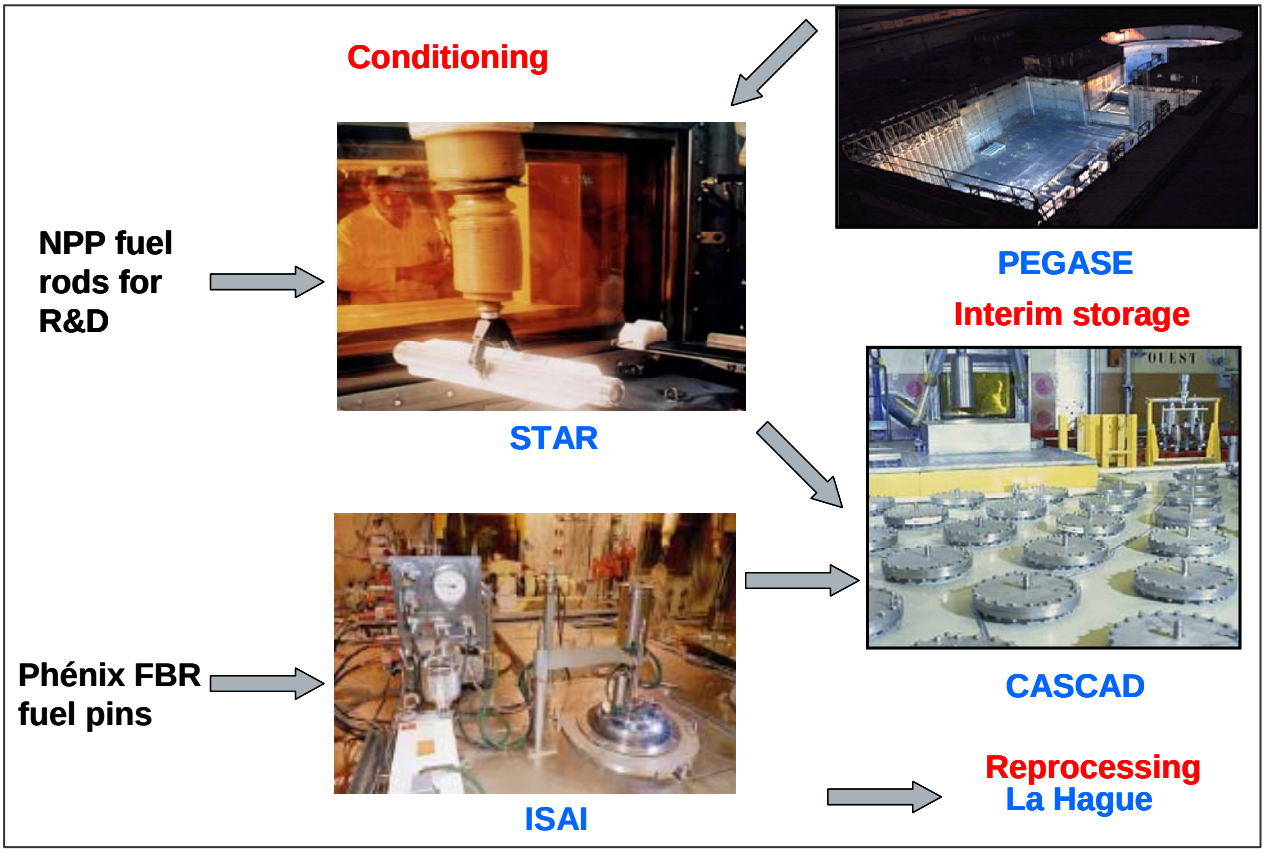

Fig 3. Conditioning and interim storage for spent fuels

Once the fuels are put in tight-welded canisters, they can be stored in devoted facilities, located in Cadarache, for spent fuel interim storage:

- PEGASE is pool-type storage. It is an old reactor pool which has been transformed to receive canister and store them under water. However, this facility does not comply with nowadays safety rules, especially for withstanding a hypothetical severe earthquake. For this reason, spent fuel canisters are removed progressively, put into new canisters in STAR and then transferred to CASCAD or the RES Canal (Fig 3).
- CASCAD is a more recent facility, where canisters are stored in dry wells.

- The RES is an experimental reactor for naval propulsion purpose, which includes a water canal in which some spent fuel canisters can be stored up to 2024.
- In Saclay, the INB 72 (INB means Nuclear Facility in French) has also some old fuels under storage in wells and pools.

Some R&D is under way to find a method to separate epoxy from fuel and enables an easier conditioning of epoxy-embedded samples from metallography.

2.2 Fissile materials

CEA has three facilities for un-irradiated fuels. MMB is used for storage of depleted & natural uranium, as well as thorium. MCMF is used for plutonium and enriched uranium. However, it is an old facility, no longer complying with safety standards, and all its fissile materials should be progressively transferred to a new facility called MAGENTA, to be opened before the end of 2010. To decrease the stockpile, large quantities of fissile materials have already been sent to La Hague for recycling.

Fig 4. MAGENTA: storage of fissile materials.

3 Facilities for radioactive liquid wastes

3.1 Aqueous liquids

Each CEA nuclear centre includes a liquid waste facility, since its start in the 1950' or 1960'. So most of them were old and needed some refurbishment when entering the 21st century.

First, these facilities need to be adapted to the current liquid waste production. This is especially true for Marcoule, there the UP1 reprocessing plant stopped its activity in 1997, so the quantities of wastes generated are decreasing sharply and depend mainly from the dismantling activity.

Second, these facilities used to embed the sludge inside bitumen. This has been performed for a very long time and today more than 60,000 bitumen drums are stored in the Marcoule interim storage. But after many years it appears that some radiolysis can occur inside the bitumen, inducing swelling, and this leads to switch now to concrete as embedding material. Concrete is also a better shielding material than bitumen, which is mainly used for confinement.

Third, these facilities needed also to be adapted to the current regulations, especially concerning improvement of liquid retention in case of accidental leak (French law issued Dec. 31, 1999) and new seismic rules for nuclear sites (issued 2001).

As the costs involved in the refurbishing of these three facilities were very high, it was decided in 2005 to build only one cementation plant for both Cadarache and Marcoule, located on the latter centre. Thus, after evaporation, concentrates from Cadarache are to be shipped to Marcoule. The refurbishment planning is as follows: STELLA-OPALE in Saclay: 2003 – 2011, AGATE in Cadarache: 2005 – 2011, AMETISTE in Marcoule: 2007 – 2015.

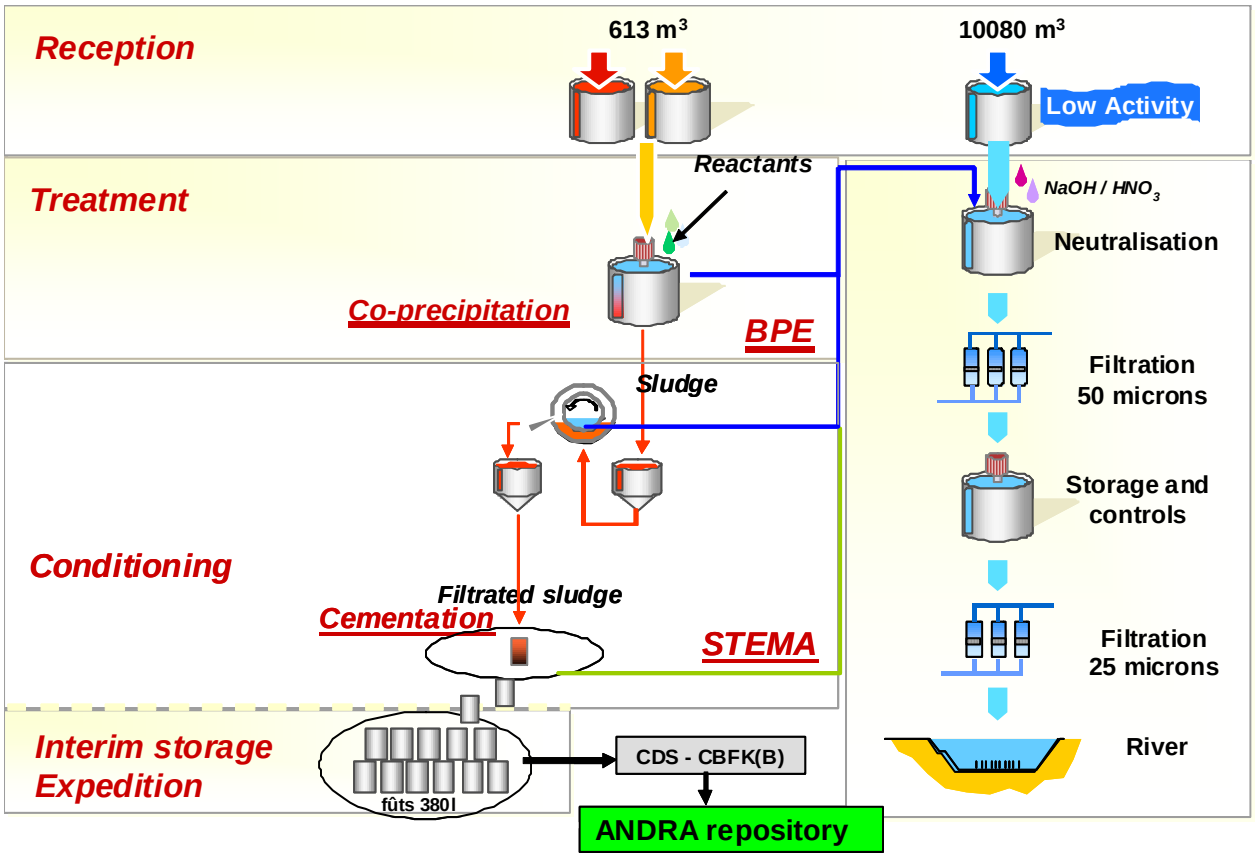

Fig 5. Schematic of liquid waste treatment in Marcoule as foreseen in 2015.

A typical refurbishment project like AMETISTE in Marcoule involves three items: decommissioning of tanks and buildings no longer needed (this will help to improve safety and decrease exploitation costs), refurbishing of builldings to comply with safety rules (typically seismic reinforcements) and new buildings when necessary (e.g. for implemanting the new cementation process).

3.2 Organic liquids

As a result from history, some volumes of organic liquids, for instance used for separation purpose in reprocessing R&D, remains stored in our facilities. These liquids represent a potential hazard, because there are both radioactive and flammable. Low contaminated ones can be sent to CENTRACO in Marcoule for burning. But for those containing too much radioactivity, another process has been developed in CEA: Hydro Thermal Oxidation (HTO), using supercritical water. HTO leads to destruction of organic molecules.

A new facility, called DELOS is under final testing in Atalante (Marcoule). The first step is purification of liquid by alkaline cleaning, to remove β, γ contamination. Then evaporation enables to send organic distillates for burning in CENTRACO. The last step is the HTO. However, implementation of such a process in hot cell is very tricky: this equipment is under pressure (250-300 bars), under temperature (500°C), inside a lead shielding (which needs cooling!) and inside a glove box (uneasy access), and the liquid flows are very small.

4 Facilities for radioactive solid wastes

Solid waste final disposal depends from the radioactivity level.

- Very Low Level Wastes can be shipped to ANDRA national CSFMA final repository,
- Low Level Wastes can be shipped to ANDRA national CSA final repository,
- Other wastes have to be stored waiting for the opening national repository by ANDRA. Underground facility is foreseen around 2025. So CEA should wait using interim storage facilities on its sites.

Here again, each nuclear centre has one or more solid waste facilities. Some are old and the safety authority requires wastes should be retrieved and stored under better conditions. New facilities are also needed to deal with wastes produced in the dismantling operations (decommissioning of Grenoble, Fontenay-aux-Roses, UP1 plant and other minor facilities).

Fig 6. Characterization of wastes (left), preparation of a box before filling with concrete (right)

For dose rates below 10 Gy/h, a new facility called CEDRA has been opened in Cadarache in 2006. It includes a medium activity storage area inside wells, and two low activity storage areas. Wastes are embedded with concrete inside steel drums. Future extensions of CEDRA are planned, when the present buildings will be filled up.

Fig 7. CEDRA interim storage: medium activity (left), low activity (right).

Another facility called DIADEM is expected by 2016 in Marcoule, to deal with wastes with a dose rate > 10 Gy/h or an alpha activity >185 GBq.

An always greater attention is given to the reduction of waste volume:

- by compaction: several facilities include a compaction press;
- by a better characterization of wastes (X-rays, gamma screening, etc…).

Another facility comprising a fusion furnace is used for recycling of contaminated lead issued from dismantling works.

5 R&D support

Some support is needed to develop new processes:

- The changes from bitumen to concrete for embedded the sludge has generated a lot of R&D tests for checking sludge compatibility with cementation. Some pilot testing remains also necessary before starting real operation on radioactive materials in the facilities.
- Some exotic materials, such magnesium cartridges from old fuels issued from graphite reactors of the first generation, need special tests for cementation.
- For liquid wastes, some improvements are tested on the filtration processes.
- Storage of alpha-containing wastes can lead to radiolysis and detrimental effects. So investigations are needed, e.g. to predict hydrogen production due to irradiation inside bitumen drums, or inside well storages, etc…
- Organic liquid destruction needs also pilot experiments.

Fig 8. Pilot mixer for concrete (left), test on magnesium cartridge (right)

6 Conclusions

CEA possesses several service facilities for treating radioactive liquid and solid wastes and the storage of spent fuels located in each nuclear centre.

- The 3 radioactive liquid waste facilities are or will be renewed and adapted to the present needs and standards.
- Two new buildings are planned for interim storage of medium activity solid wastes: CEDRA (in operation, with possible extensions) & DIADEM (planned for 2016).
- One new building for interim storage of fissile material (MAGENTA) will start this year.
- R&D is necessary to support these projects (cementation process, radiolysis, etc.).

7 Reference

[1] Presentation of CEA hot laboratories, with a focus on LECI, J.-Y. Blanc, IAEA TCM on Hot Cell Post irradiation Techniques and Poolside Inspection of Water Fuel Assemblies, Buenos Aires, 27-30 Nov. 2006.

PREPARATIONS & EXAMINATIONS NECESSARY FOR OBTAINING A LICENCE FOR TRANSPORTING RADIOLYSABLE MATERIAL

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ABSTRACT

The Phébus FP project is an international project in the field of severe accidents that is performed by IRSN in the Phébus facility operated by CEA on the site of Cadarache in France and involves many international institutes as well as the European Commission. It consists of a series of tests in which a 20 fuel rods bundle of 1m-long (18 pre-irradiated UO_2 fuel rods and 2 fresh fuel rods) was subjected to overheating (to simulate the loss of coolant) under various conditions and allowed to degrade and form a molten corium pool. The FPT1 test bundle (a test under flowing steam conditions) was transported to ITU to perform post – irradiation examination (PIE) at different locations. The bundle was stabilised by filling with epoxy resin to enable sectioning. This was followed by considerable microscopy observations & analysis of the bundle. After the experimental campaign had been completed, it was necessary to return the bundle to CEA at Cadarache. However, the licensing authorities did not permit such transport in the casks used to bring the bundle to ITU due to possible risks associated with the radiolysis of the embedding resin (and consequent hazardous gas generation) caused by the elevated radioactivity of the irradiated material.
This presentation details the gas analysis carried out by ITU for CEA that enabled accurate radiolysis estimates to be made and submitted to the French authorities and transport company. The outcome of this campaign of measurements allowed proposing acceptable conditions for the FPT 1 bundle return transport.

1 Introduction

The Phébus FP project is an international collaboration in the field of severe accident research in which five tests were carried out degrading a 20 rod-bundle with 1m long rods under various conditions. The project was managed by IRSN and supported by the European Commission along a series of EU national institutes as well as non-EU research institutes or authorities [1]. The bundle was severely overheated under different conditions in each test and degradation of the bundle and the release of the fission products into a simulated primary circuit and containment was monitored. The FPT1 test was carried out in July 1996 and was the degradation of an irradiated UO_2 fuel bundle (23 GWd/tU mean burn-up) under high steam flow conditions [2]. ITU was contracted to do the sectioning & ceramographic examination of the FPT1 bundle.
The FPT1 bundle was embedded in resin and sent to ITU for sectioning and macroscopic photography and then microscopy and analysis on selected parts. The work was finished

with an ITU report in Oct. 99 (JRC-ITU-TPW-99/31) and a final FPT1 report was issued by IRSN in Dec. 2000 [3]. Since this time IRSN has attempted to organise the transport as foreseen in the original contract but one of the major problems had been that the proposed intermediate storage facilities belonging to the CEA (Pègase) no longer accepts the bundle. As it contains epoxy resin it is assumed that it could generate hydrogen through radiolysis of the resin and present a fire hazard. Moreover, it would not be possible to carry out any transport without a special licence for the current transport casks again because of the presence of the radiolysable material.
In discussions with the CEA, they contracted ITU to carry out measurements on the possible hydrogen release through radiolysis on a portion of the bundle material in order to quantify the risk and so present a more substantiated case to the licensing authorities for permission for the transport.

2 Testing

2.1 Container Construction

ITU constructed a special container in stainless steel 12.5cm diameter and 25cm high with a sealed airtight cover with the possibility of sampling from the container. The container was tested to 5.5 bar in He atmosphere for 48 h and was found to be airtight (<1.0% loss of pressure). The vessel is shown in Fig. 1.

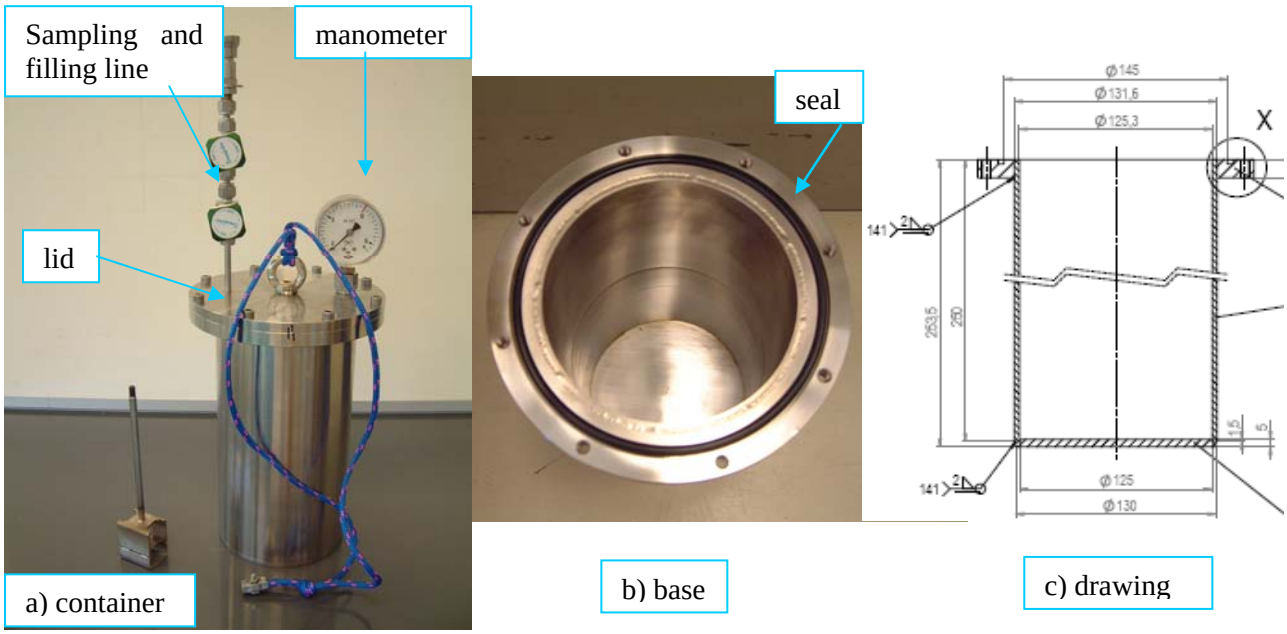

Figure 1 Stainless steel container (air-tight to 5.5 bar) constructed for testing of the H_2 generation from FPT1 segments.

2.2 Selection of discs

A selection of discs was taken from the storage containers in ITU's storage cell. The discs came from a range of positions including the upper bundle, the central cavity, the corium pool and the lower bundle so that all conditions and ratios of resin to fuel or melt were included. The discs selected are given in Fig. 2. The total height was 21.8 cm and so represented about one–fifth of the total bundle. The resin surfaces of the discs did not show any degradation and handling of them did not cause any crumbling.

2.3 Loading Procedure

These discs were loaded into the container which was then closed and sealed by screwing the 6 bolts down firmly. The connections were then used to evacuate the gas and to back-fill with Ar (5.0 purity - Air Liquide). This was repeated three times so as to ensure that the atmosphere contained no prior impurities. The final gas pressure was above atmospheric in order to be able to withdraw future gas samples (2.2 bar gauge). A gas ampoule was attached and filled using the over-pressure from the gas, before evacuating and refilling from the container. After repeating this 10 times, it was filled with gas that would be the initial

sample for the test. It would provide background levels of impurities prior to the start of the test. The temperature in the hot cell B104 where the container was placed was between 25 and 28°C.

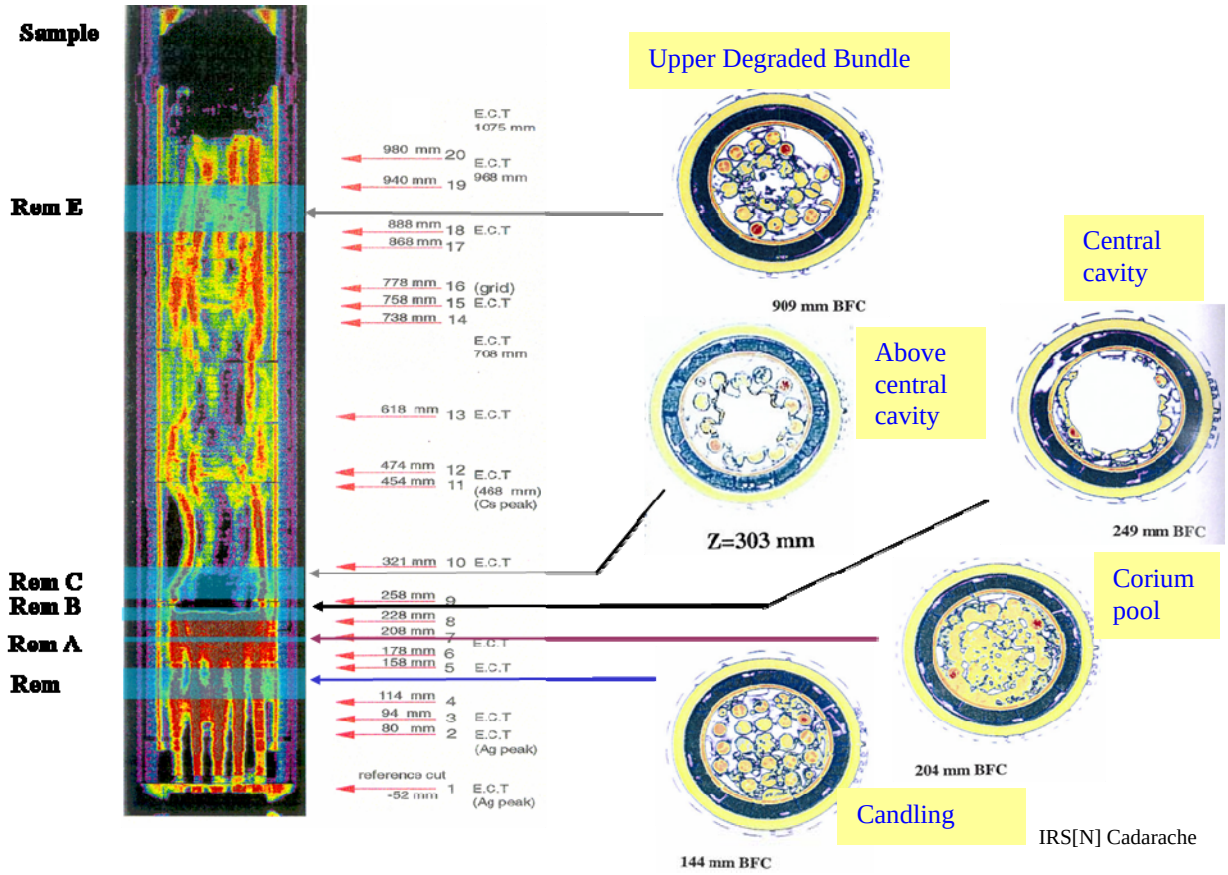

Figure 2: Diagram showing the FPT1 bundle tomography and the samples (or remnants) selected for radiolysis testing along with tomographies of their cross-sections that were used to estimate the percentage of resin in the selected discs in Table 1 (Sectional Tomography carried out by IRSN)

2.4 Test Procedure

After 69 days (start date: 15th August '07, final sampling date: 23rd Oct. '07) a second gas ampoule was attached to the container and using similar procedure (5 times filling and evacuating of ampoule and gas lines), the gas from the container was sampled using the ampoule. The overpressure in the container ensured efficient sampling. The samples were moved to the next hot cell where they were connected to the gas line with a direct connection to the gas mass spectrometer (Balzers Omnistar MS). The Mass Spectrometer was purged with the gas from the ampoule and evacuated 10 times before a sample was taken for analysis. The mass spectrometer was calibrated with three certificated gases (from Air Liquide AG). These were 1 ppm H_2 in Ar, 100 ppm H_2 in Ar and 1% H_2 in Ar; these gases were chosen so as to enable accurate measurement at low levels.

2.5 Testing

The measurements of the samples along with these calibration gases were then carried out. Figure 3 shows the initial and the final gas spectra from the samples. It is seen that the main peaks come from the Argon. Thereafter H_2, N_2 and CO_2 are present with low hydrocarbon residues. Possibly some Cl traces may be present. The estimated overall composition is Ar ~ 97%, H_2 ~ 2%, 0.1% < N_2 < 1%, 0.05% < CO_2 < 0.5%. The N_2 may be due to slight leaks in the system or could point to the presence of functional groups such as amides in the resin or the hardener. The hydrogen content in the second sample taken from the container based on the calibration with certificated gases was estimated at 19,200ppm + 20% (from all sources of error). Therefore a value of 2% +0.4% (absolute) ie 1.6% to 2.4% was

determined. The absence of fission gas (Kr & Xe) traces was noted, so that the fuel appears to be stable.

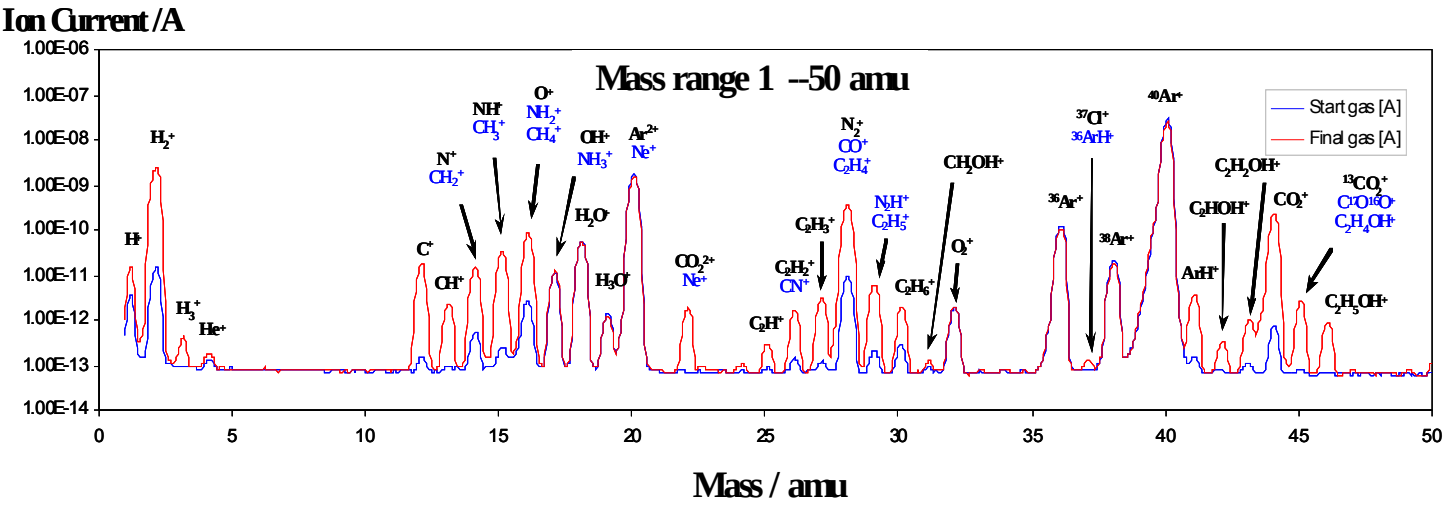

Figure 3: The initial gas mass spectrograph (in blue) and the final spectrum (in red) taken after 69 days. In addition to the Ar peak, some hydrocarbon residues, N_2, O_2 and H_2 are seen.

3 Calculation of the Resin Volume and Fuel Mass in the Sample & Bundle

The positions and thicknesses of the discs cut from the bundle and the remnants from the FPT1 bundle are given in Table 1.

	remnants			discs			
Segment	**upper position**	**lower position**	**thickness/mm**	**Zone**	**upper position**	**lower position**	**thickness/mm**
remnant H	6.5	78.2	71.7	Lower candled rods			
Disc 14				Lower candled rods	82.3	96.5	14.2
Disc 4				Lower candled rods	98	118	20
Rem -	118	160.2	42.2	Lower candled rods			
Disc 3				corium pool	160.2	178	17.8
Disc 2				corium pool	179.3	194.5	15.2
Rem A	194.5	208	13.5	corium pool			
Disc 1				corium pool	208	228.9	20.9
Rem B	228.9	249	20.1	corium pool/cavity			
Disc 5				Cavity	248	261	13
Rem C	261	323	62	Cavity			
Disc 6				Cavity	323	343.2	20.2
Rem F	343.2	452.7	109.5	Cavity			
Disc 12				Cavity	452.7	472.7	20
Rem I	472.7	588.7	116	Upper cavity			
Disc 13				Upper cavity	588.7	606.4	17.7
Rem J	606.4	739.5	133.1	Upper bundle			
Disc 7				Upper bundle	739.5	757	17.5
Disc 8				Upper bundle	758.5	776.4	17.9
Rem D	776.4	867.4	91	Upper bundle			
Disc 9				Upper bundle	867.3	886	18.7
Rem E	886.1	938.6	52.5	Upper bundle			
Disc 10				Upper bundle	938.6	958	19.4
Disc 11				Upper bundle	959.5	979	19.5
Rem G			88.5	Resin			
		total	**800.1**			**total**	**252**
		% remnants	**76.05**			**% discs**	**23.95**
					Total remnants & discs/mm		**1052.1**

Table 1 Positions and thicknesses of all examined discs (disc 1-14) and the remnant sections (Rem- to Rem J) in mm of bundle height. The zone to which the various sections belong is also given.

The calculations for the resin volume and the melt (fused fuel, cladding and structural material) for each disc are based on the cross-section of the 20 rod FPT1 bundle with a central Ag-In-Cd control rod combined with the thicknesses of the discs. Table 2 gives details of the estimated partition between resin and fuel for each of the discs used in the test. It also gives notes of the bundle's and fuel rod's dimensions as well as the calculations of the cross-sections of each area. Thus Rem C as part of the central cavity was estimated as having only an 8 mm wide thick outer ring of material with a10 % porosity in order to calculate the percentage of cross-section as melt and that of free volume filled with resin. By contrast Rem E from the upper bundle was calculated as having a cross-section equivalent to having one rod extra in cross-section for half its height. Remnant sections were selected as they were still intact and had not been used for microscopic examination.
Five sections for the test were chosen that came from each of the major zones of the FPT1 bundle: Rem (candling rod zone beneath the corium pool), Rem A (middle of corium pool), Rem B (upper corium pool and cavity), Rem C (central cavity), Rem E (upper degraded bundle).The central cavity is estimated to start at 240 mm, therefore Rem-B (ca. 230-250 mm) is taken as 50% cavity and 50% corium pool. No sample was taken from the section above the bundle where only resin was present.

		Partition		**Linear mass** (see Figure 4)	**Calculation Notes**
Zone	**Remnants**	**fuel %**	**resin %**	**g/mm**	
Lower candled rods	Rem	40	60	20.5	4)
corium pool	Rem - A	95	5	26	5)
Cavity/corium pool	Rem - B	65	35	17	6)
Cavity		36	64	8.5	7)
Upper cavity	Rem -C	20	80	12	8)
Upper bundle	Rem -E	34	66	15	9)
Resin		20	80	2	10)

Calculation Notes

Dimensions of bundle

1) Thoria sheath 73 mm int. dia., fuel cladding ~9.5 mm ext. dia.
2) Starting configuration: 20 rods, 1 control rod and guide tube (ext. dia. 12.1 mm) & 2 ultrasonic thermometers – UTS - (ext. dia. 6 mm).

Details of calculations

3) The ThO_2 cross-sectional area is $\pi(73/2)^2 = 41.854$ cm² and that of the rods is $[20.\pi(9.5/2)^2 + \pi(12.1/2)^2 + 2.\pi(6/2)^2] = 15.891$cm² , the difference is 25.96 cm² which gives 62.0% of the cross-section will be filled with resin for a 'full' bundle.
4) Rem assumed to be a full bundle but with a blockage on the south side equal to one extra rod (% extra section $\pi(9.5/2)^2/ \pi(73/2)^2 = 1.7\%$. Subtract 1.7% from 62.0% to give 60.3% of the cross-section filled with resin.
5) Rem A is assumed to be corium with residual resin in open pores (estimated at 5%)
6) Rem B is taken as 50% corium pool (similar to Rem A) and 50% Cavity, and so average values were taken of linear mass 17 g/mm and 35% resin content.
7) The Cavity is assumed to have an empty centre with only 8.1 mm thickness of material at the edge (r/r_0= (36.5-8.1)/36.5=0.78. Hence $1-[r/r_0]^2 = 0.394$). This edge has 10% porosity, hence 39.4% x 0.9= 35.5% of the section is fuel & 64.5% of section is resin
8) Rem C is taken as 12 whole rods in section $(\pi(73/2)^2-12.\pi(9.5/2)^2 = 33.34$cm² or 80% section resin-filled.
9) Rem E is taken as 20 whole rods $(\pi(73/2)^2-20.\pi(9.5/2)^2 = 27.67$cm² or 66% section resin-filled.
10) This section was estimated from tomography to have a 80% resin-filled section

Table 2 estimates of resin volumes (as % of cross-section) using the assumptions given in 'Details of calculations'.

In Fig. 4 the FPT1 bundle is displayed with the discs selected along with the linear bundle mass/mm along the bundle's axis (taken from the FPT1 final report [3]). The linear mass in g/mm is then read off for the selected discs. Using the disc's height the bundle mass can then be estimated for each selected disc.

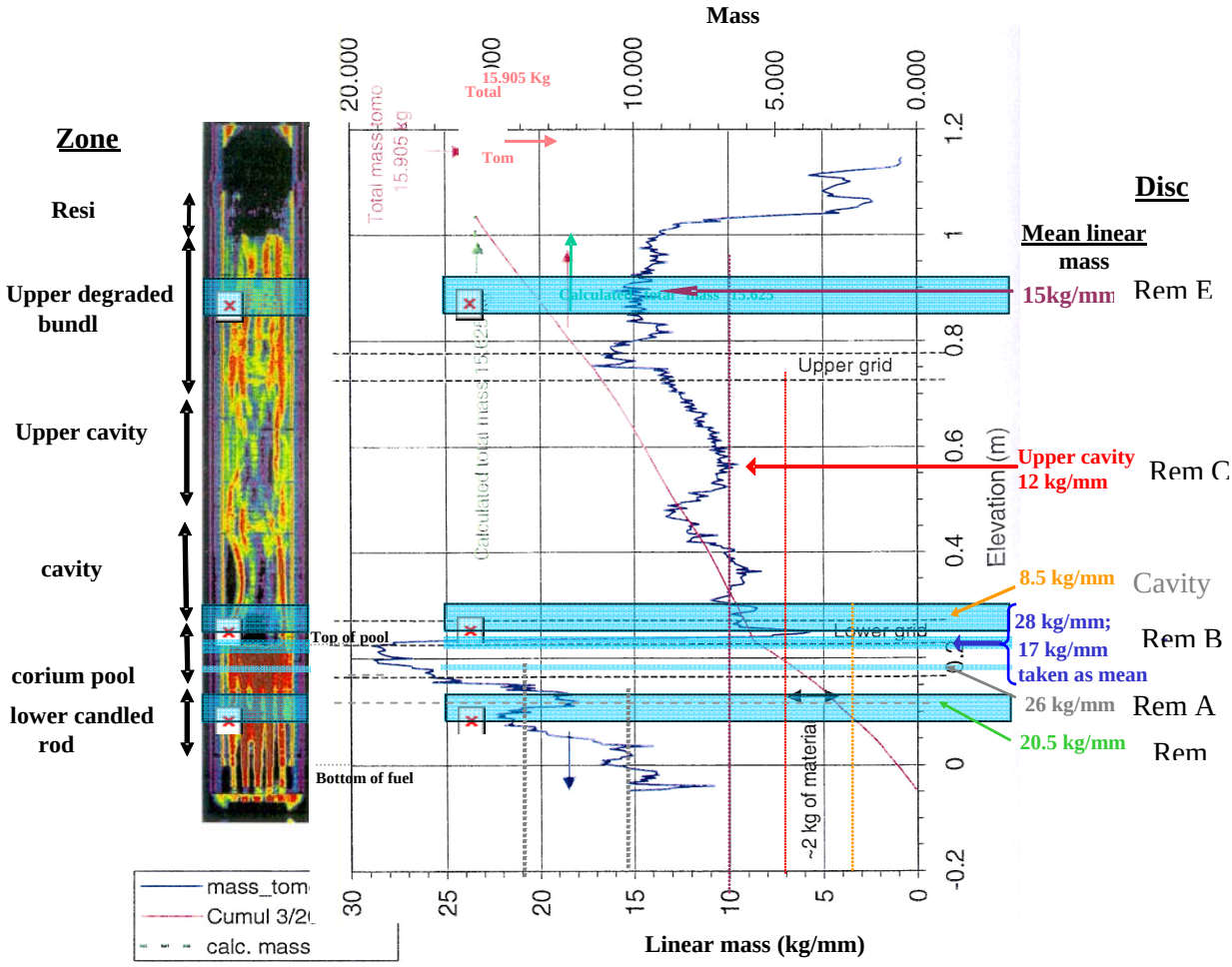

Figure 4 Diagram giving the linear masses of the various zones of theFPT1 bundle that have been selected for the samples (& given in Table 2) and used to calculate the disc's mass in Table 3

The estimate of volume percent of resin and the mass of reactor material for the selected disc from a particular zone was taken as typical of all samples from the same zone. These values are calculated and given for all discs and remnants of the degraded FPT1 bundle in Table 3.

Segment	Zone	Mass/kg fuel etc	Quantity/cm³ resin
remnant H	Lower candled rods	1,470	180,06
Disc 14	Lower candled rods	0,291	35,66
Disc 4	Lower candled rods	0,410	50,22
Rem -	Lower candled rods	0,865	105,97
Disc 3	corium pool	0,463	3,72
Disc 2	corium pool	0,395	3,18
Rem A	corium pool	0,351	2,83
Disc 1	corium pool	0,543	4,37
Rem B	corium pool/cavity	0,342	29,44
Disc 5	Cavity	0,111	34,82
Rem C	Cavity	0,527	166,08
Disc 6	Cavity	0,172	54,11
Rem F	Cavity	0,931	293,31
Disc 12	Cavity	0,170	53,57
Rem I	Upper cavity	1,392	388,40
Disc 13	Upper cavity	0,212	48,89
Rem J	Upper bundle	1,997	367,67
Disc 7	Upper bundle	0,263	48,34
Disc 8	Upper bundle	0,269	49,45
Rem D	Upper bundle	1,365	251,37
Disc 9	Upper bundle	0,281	51,66
Rem E	Upper bundle	0,788	145,02
Disc 10	Upper bundle	0,291	53,59
Disc 11	Upper bundle	0,293	53,87
Rem G	Resin	0,177	296,33
	TOTAL	14,366	2771,9
	Discs 1 to 14	4,162	545,5
	Rem +Rem A to J	10,203	2226,5

Total in Container (Rem & Rem A,B,C & E)	
Quantity/kg fuel	Quantity/cm³ resin
2,872	449,34
Ratios resin /fuel mass cm³/kg	
overall	192,96
tested discs	156,44

Table 3 calculation of the mass of nuclear melted material (in kg) and resin volume (cm³) for all FPT1 bundle segments with those selected for radiolysis measurements highlighted in yellow. The ratio of resin volume/melt mass is also given for the whole bundle and the selected segments.

Table 3 gives the resin volumes & melted material masses for each section and the totals for the FPT1 bundle and the tested sections. Approximately 3kg of fuel, cladding & structural material along with 450cm³ of resin is in the sample, this compares with 14.4 kg of fuel & material and 2770cm³ of resin for the whole bundle. The resin volume/material mass for the radiolysis test sections and for the whole bundle is also given: these are 189 cm³/kg in total and 138.9 cm³/kg for the radiolysis test. These values are comparable and so suggest that the sample is representative of the whole bundle.

The free volume in the container was calculated as the internal volume of the container minus the volume of the discs. The height of the discs in the container was found to be 3.2 cm below the lip of the container (ie 21.8 cm height) this was higher than expected from the cuts (19.2 cm), and this is presumed to come from the inaccuracy in the cut positions. Nevertheless the estimate of the evolved hydrogen is given in Table 4 which shows that 2% H_2 content was found in a free volume of the container of 920.2 cm³. This is a total production of 18.4 cm³ after 69 days. This, in turn, yields for the whole FPT 1 bundle with approx. 2750 cm³ of resin, a daily H_2 production rate of 1.65 ±0.33 cm³/day (ie from 1.32 to 1.98 cm³/day including all sources of error) based on resin mass. If the calculation is based on fuel & material mass (ie dose rate) then this gives a value of 1.33cm³ H_2/day. .However the value based on resin volume was taken as more conservative & possibly more accurate.

Disc & Rem. Vol. in container/cm³	2147,74	
Container volume /cm³ (25 cm long x 12.5cm dia)	3.067,96	
Free volume/ cm³	920,22	
Gas MS measurement of H_2	**2%**	absolute error ± 0.4%
Total Gas release -cm³ in 69 days (15th August-23th Oct.'07) for 449.34 cm³ resin and 2.87kg fuel	**18,40**	3,68
Total daily gas release -cm³ H_2 /FPT1Bundle/day	**1,65**	0,33

Table 4 showing the estimated gas release for the selected discs and the estimate for the FPT1 bundle

4 Conclusions

1) Measurements have been carried out on the gas composition in a sealed container containing 22cm high segments, representative of the various parts of the degraded bundle. Gas mass spectroscopy has shown a concentration of H_2 of 2% (± 0.4% absolute) has built up in the container in 69 days. Some N_2 and some hydrocarbon residues were also detected.
2) This level of 2.0% gives a total volume of hydrogen released from the resin during the 69 day test of 18.40 ±3.68 cm³.
3) This gives an estimate of approximately 1.65 ± 0.33 cm³ of hydrogen per day for the whole FPT1 bundle based on resin mass. This is a negligible production rate and would represent no risk for the purposes of a transport considering its short duration and that such large transports of nuclear material use airtight containers with an inert atmosphere.
4) No fission gas releases were measured so that the fuel appears to be stable.

5 Postscript

This data was transmitted to the transport company and the licensing authorities and they accepted this data in giving a licence for the transport of this material containing radiolysable material in the standard container. However additional matters needed clearing before the transport will take place. Firstly, not only the transport container (a standard requirement in Type B containers containing Pu material) but also the inner containers must be capable of evacuation of the atmosphere and back-filling with inert gas. Secondly, licence for storage at CEA was being applied for at the same time as the transport licence. Here they required that there would be no build-up of the pressure in the inner containers. Thus it was decided to make outer containers for the samples with small apertures and inner containers with poral filter apertures in order to avoid any pressure build-up. This design, even with fine metallic filters where a very small risk of slight contamination was possible, was preferable to the authorities to using airtight containers with the risk of some (albeit small) pressure build-up for longer storage conditions.

6 References

[1] P. von der Hardt, A. V Jones, C. Lecomte, A. Tattegrain, Nuclear Safety Vol **35** (2), 1994.
[2] H. Scheurer, & B. Clement, PHEBUS FP Data Book FPT1, Document PH-PF IS/92/49, May 1997
[3] D. Jacquemain, S. Bourdon, A. De Braemaecker and M. Barrachin, Phébus PF – FPT1 Final report (IPSN Tech. Report No. IP/00/479) Dec. 2000.

TEN YEARS OF EXPERIENCE IN THE C11/C12 SHIELDED LINE OF THE ATALANTE FACILITY AT MARCOULE

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ABSTRACT

The C11/C12 shielded line built in the late 1980s and hot commissioned in 1999 is reserved for high-activity R&D on spent fuel reprocessing and particularly for dissolution studies. The programs undertaken since the startup have required the installation of suitable experimental or analytic equipment. Other upgrades were necessary to comply with new operating constraints, to address the obsolescence of some instrumentation and control devices, or to replace equipment subjected to accelerated aging due to high dose rates and the presence of corrosive solutions. The lessons learned from ten years of operating experience in this line are discussed, and the upgrades implemented to maintain suitable safety and quality levels are described together with the modifications under consideration for future programs.

1. Introduction

The C11-C12 shielded line installed in the CEA's Atalante complex at Marcoule is dedicated to research and development on the high-activity chemistry of spent nuclear fuel. The programs carried out since hot commissioning in 1999 have required changes in the shielded line to add suitable experimental or analytic equipment. Other upgrades were necessary to comply with new operating constraints, to address the obsolescence of some instrumentation and control devices, or to replace equipment subjected to accelerated aging due to high dose rates and the presence of corrosive solutions. The lessons learned from ten years of operating experience in this line are discussed, and the upgrades implemented to maintain suitable safety and quality levels are described together with the modifications under consideration for future programs.

2. Programs completed since hot commissioning

The work carried out in the ten years since hot commissioning concerns three major programs:

Confirming the scientific feasibility of enhanced separation of the minor actinides recovered by the PUREX process, which was demonstrated in 2005.
Spent fuel dissolution studies.
Characterization or process studies in support of industrial operation by AREVA, such as studies of industrial hulls, precipitates, and deposits.

2.1 Enhanced separation studies

One strategy for separating the minor actinides is to coextract them together with the lanthanides from a PUREX raffinate by the DIAMEX process, and then separate the lanthanides by the SANEX process (**Fig** 1. Actinide separation flowsheet

).

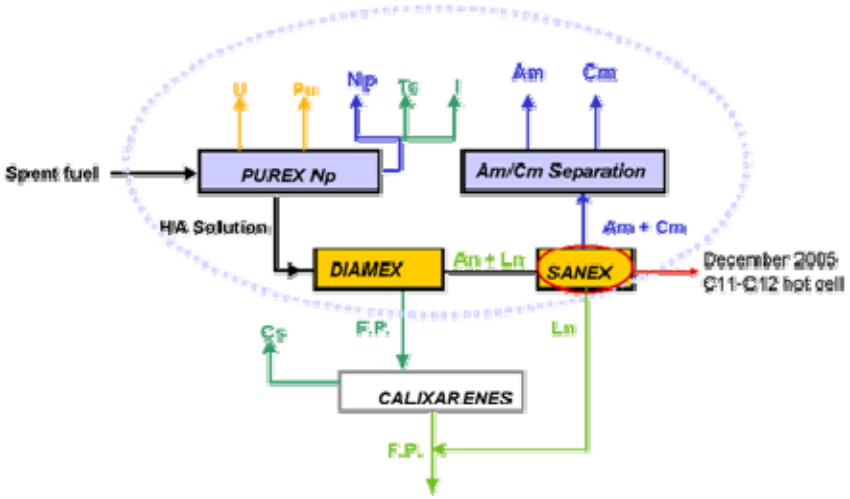
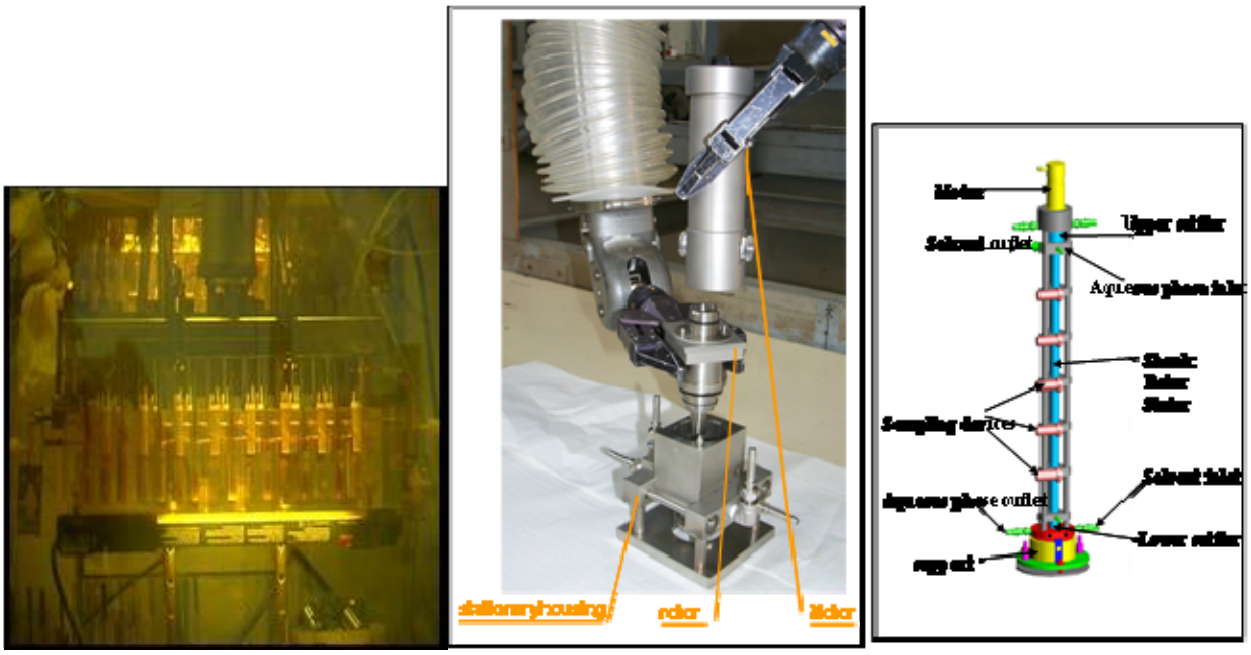


Fig 1. Actinide separation flowsheet

The implementation of a process for enhanced separation of the minor actinides required the design and development of high activity liquid-liquid extraction devices (**Fig** 2.) including mixer-settlers, centrifugal extractors, and Taylor-Couette columns. These devices designed for use in high-activity environments via telemanipulators are used to extrapolate the performance of the test processes to industrial scale.



Mixer-settler **Centrifugal extractor** **Taylor-Couette column**

Fig 2. Liquid-liquid extraction devices

They were used to demonstrate the scientific and technical feasibility of several enhanced separation processes for the minor actinides, including the DIAMEX-SANEX process allowing separation from a PUREX raffinate. This test was successfully completed in 2005 with the separation of more than 99.9% of the americium and curium from a PUREX raffinate.

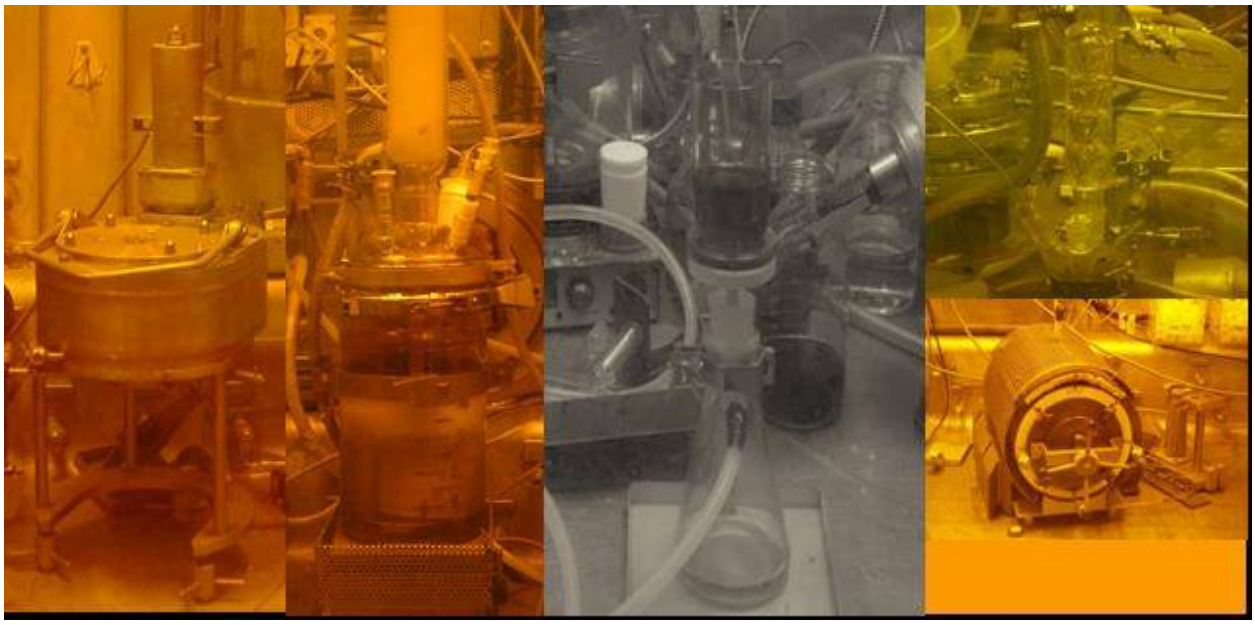
2.2 Dissolution studies

Spent fuel dissolution studies mainly concern:

UOX fuel irradiated in boiling water reactors (BWR) or pressurized water reactors (PWR) with burnup values ranging from 30 to 70 GWd/t,
MOX fuel after 2, 3 or 4 cycles,
research reactor fuel (UMo, USi),
actinide targets irradiated in a fast-neutron reactor (Phenix).
The parameters studied were the type of fuel and its burnup, the dissolution temperature, the cutting pitch of the dissolved segments and the dissolution medium (in acid or by melting in molten salts).

Fuel reprocessing includes the following steps (**Fig**. 1):

The fuel rods are cut into segments 15 to 40 mm long by a smooth-cut rotary shear to minimize the cutting swarf. A special shear has also been designed to cut plate-type UMo and USi research reactor fuel. The dimensions of irradiation targets make them incompatible with the shear; for this reason and to limit pollution, an electron-discharge machining system was developed to open them and recover the powder for processing.
Dissolution is performed at temperatures between 70°C and the boiling point in a nitric acid medium in glass reactors equipped with cooling and nitrous fume recombination columns.
Dissolution residues are separated by filtration on cellulose filters of different porosities (8 to 0.3 µm).
The dissolution residues are calcined and dissolved by further attack in a hydrofluoric acid or molten salt medium.
Process parameters (temperature, pressure, stirring speed, corrosion potential, etc.) are measured by a computerized online measurement system; online ^{85}Kr measurement is used to monitor the progress of the dissolution. Chemical and spectroscopic analysis are performed to accurately characterize the resulting solutions.



Shearing **Dissolution** **Filtration** **Treatment of insolubles**

Fig. 1. Stages of dissolution studies

2.3 Characterization studies

Characterization studies mainly concern support for industrial operation by AREVA in the La Hague reprocessing plant. The objects characterized since hot commissioning include industrial cladding tubes (hulls), as well as deposits and precipitates that obstruct the process head-end devices.

2.3.1 Characterization of hulls

Characterization has three main objectives:

Validate measurements at waste package online characterization stations.
Measure the impact of reprocessing on waste variability.
Establish the inventory of FP and actinides in hulls after rinsing.
The batches of hulls analyzed in the facility since hot commissioning were sampled from the former UP2-400 plant and from the currently operating UP2-800 and UP3 plants at La Hague. The reprocessed fuels include UOX1, UOX2, UOX3, and MOX with Zircaloy cladding.

Hull characterization involves the following steps:

non-intrusive characterization:
quartering of large fragments, and particle size grading of the fines,
sorting of fuel assembly structural elements,
dimensional analysis of the hulls,
active and passive neutron counting in the COQUINA cell (**Fig. 2**),
intrusive characterization involves rinsing and fully dissolving the hulls and fines to quantify the radioactive contaminants and to determine their distribution and retention mode.

Fig. 2. COQUINA counting cell

2.3.2 Characterization of precipitates

The precipitates were studied with the following objectives:

characterize the deposits that obstruct the process equipment,
understand their formation mechanisms to modify plant operating parameters and prevent or limit their formation,
specify rinsing reagents and develop a chemical cleaning procedure.
A few hundred grams sampled at various points in the facilities were analyzed. Characterization involved the following steps:

Non-intrusive characterization:
general appearance,
dose rate,
particle size analysis,
Intrusive characterization:
chemical and radiochemical characterization after complete dissolution,
electron microscope observation,
testing of reagents for remedial treatment.

3. Results of operating experience

Devices with design defects or which did not comply with changing safety and security requirements had to be modified or replaced. After ten years of operation, difficulties related to aging began to appear. This section discusses the difficulties encountered in the course of these experiments.

3.1 Tank level measurement

New safety rules concerning the thermal and radiolytic hazards of dissolution solutions were necessary to improve the monitoring of storage tanks beneath the workbench in each hot cell. The tanks were fitted with temperature measurement provisions, the accuracy of the level sensors was improved, and air scavenging was implemented in the tank atmosphere to prevent any accumulation of radiolytic hydrogen. The existing level measuring probes using an imprecise capacitive measurement technique were replaced by ultrasonic level sensors (**Fig. 3**) equipped with temperature sensors and perforated to allow the passage of the scavenging air stream.

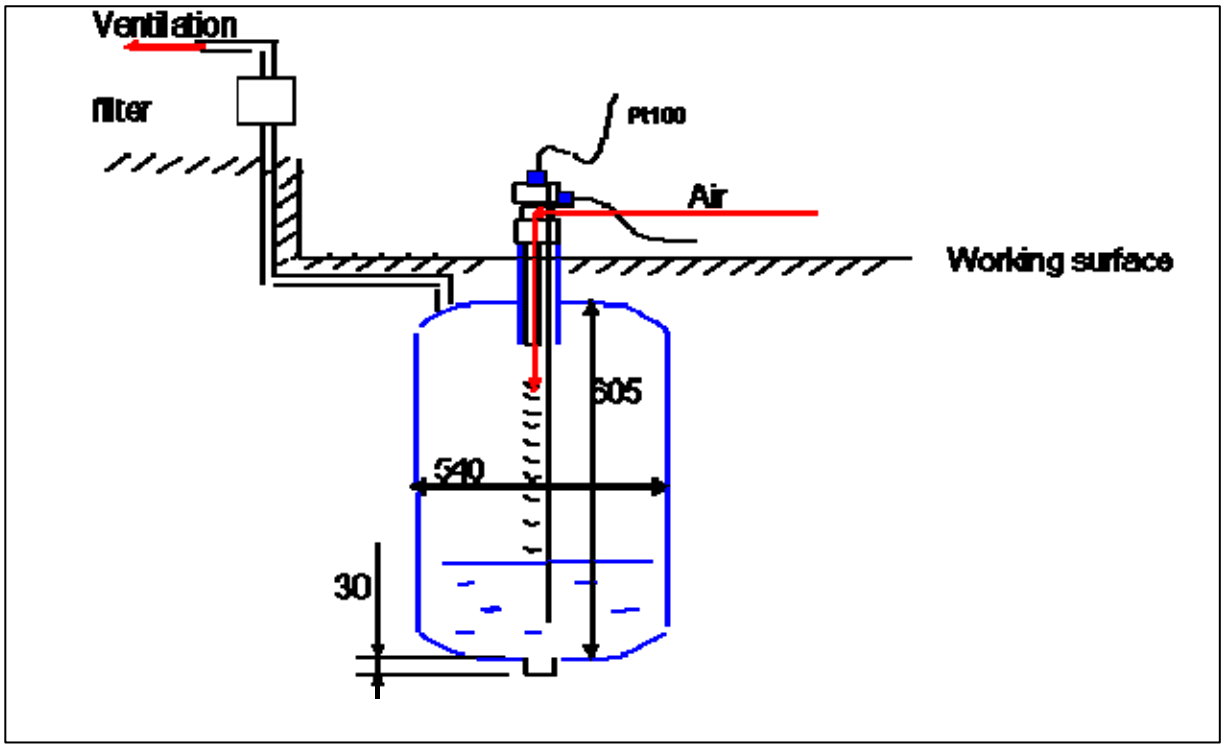


Fig. 3. Tank level and temperature measurement

3.2 Cooling the dissolver condensing columns

The insoluble gases released by fuel dissolution are washed and transferred to the shielded line ventilation system. Evaporated water and acid are condensed in water-cooled condensing columns and returned to the dissolver to avoid concentrating the solution. The water in the cooling system was originally cooled by a freon thermosiphon and two freon-water heat exchangers (**Fig. 4**). This system proved difficult to use because of problems in priming the thermosiphon. A new cooling device based on the vortex principle, capable of producing a cooled airflow from compressed air, together with an air-water heat exchanger inside the hot cell were used to cool the columns. This easy-to-use, lightweight, and maintenance-free device is now installed at all the work stations of the dissolution line.

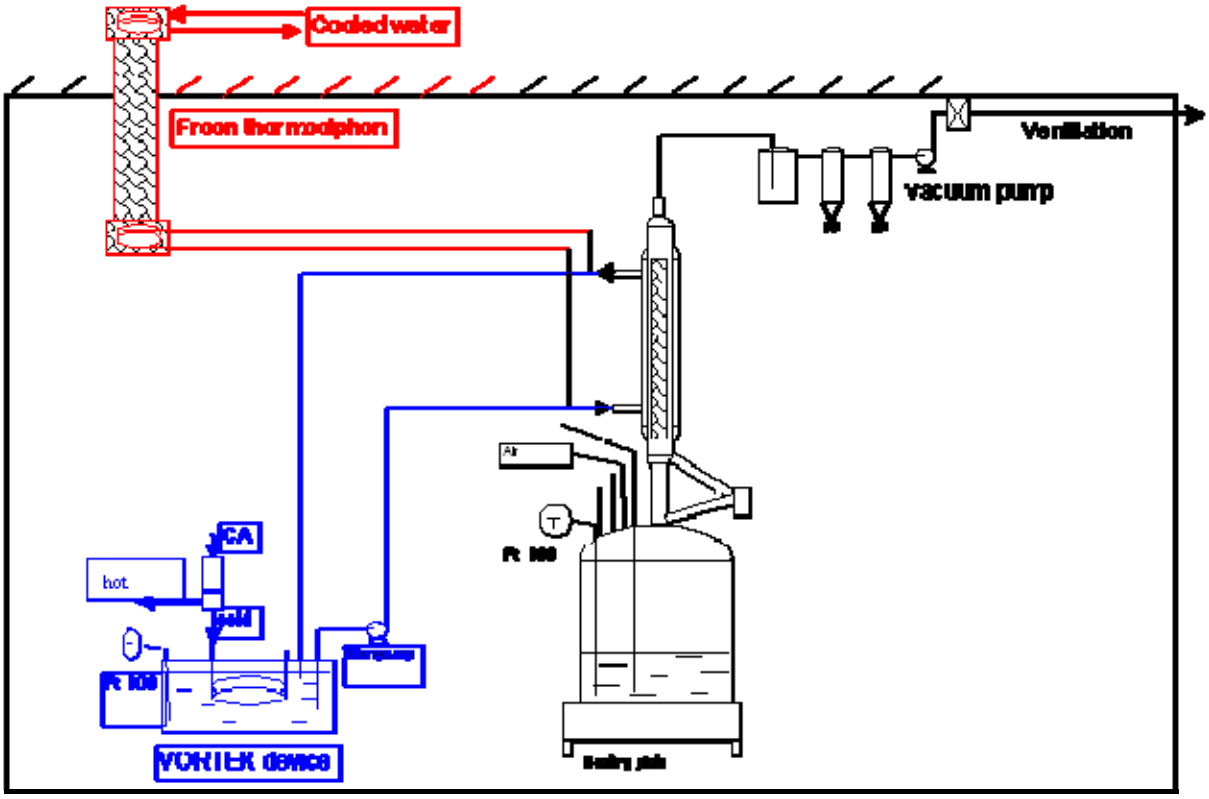


Fig. 4. Dissolution system schematic

3.3 Malfunction due to aging

The main difficulty arising from aging of the equipment concerns the electrical connector panels (**Fig. 5**). Each hot cell in the shielded line is fitted with two connector panels with about 20 electrical connectors each, supplying both power and instrumentation & control functions. An inventory of the remotely actuated Jupiter connectors in 2010 showed that 25 to 30% of the connectors were unusable due to two types of anomalies: mechanical wear of the electrical sockets, and corrosion between electrical pins resulting in spurious contacts.

Enough connectors are still available to carry out normal experiments, but the increase in the number of faulty connectors could become a problem in the future. Replacement of the electrical penetration is possible, but would be a complex operation requiring the shutdown of all experiments for several months.

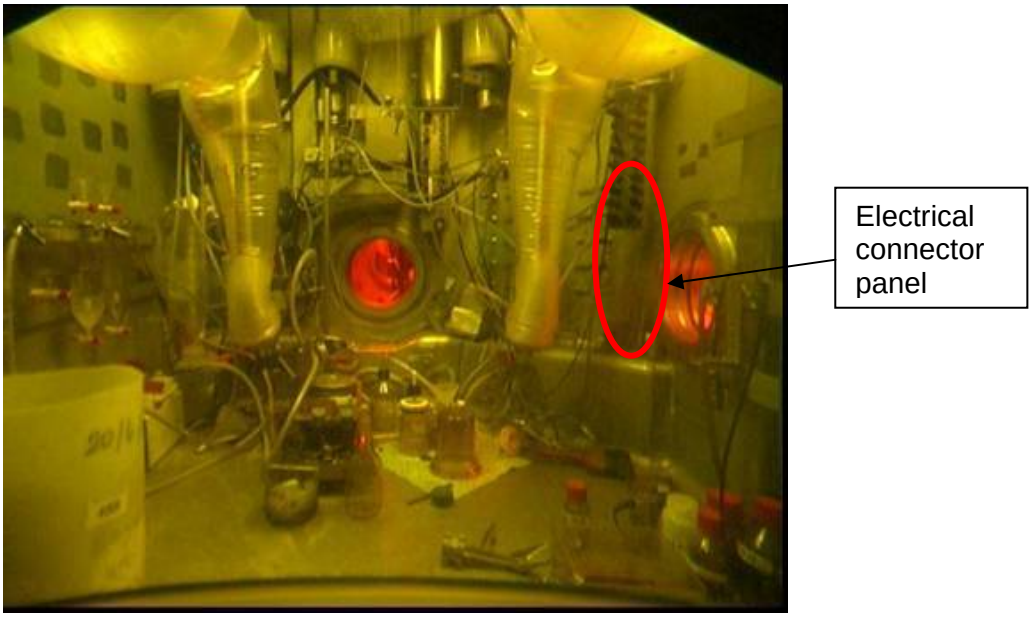


Fig. 5. Electrical connector panel in a hot cell

4. Outlook

4.1 New programs

Future programs will address two main topics:

dissolution studies for special fuels:
high burnup UOX or MOX fuel,
studies of corrosion during fuel dissolution,
carbide, nitride, or silicide fuels for future research reactors,
phenix fast neutron reactor fuel and further studies on actinide and iodine targets irradiated in a fast neutron spectrum.
characterization of industrial objects:
industrial hulls from reprocessing fuel with M5 cladding,
industrial hulls from reprocessing MOX fuel,
precipitates and deposits collected after industrial reprocessing campaigns with special fuel (MOX, UOX3, etc),
industrial dissolution fines and residues.

4.2 Hardware developments and new analytical techniques

The new programs will require the design and development of new high-activity equipment and analysis techniques.

The main changes concern the ongoing development of an ozone dissolution process that is expected to be deployed in C11 in 2011. This technique will allow complete dissolution of the fuel in a single step. For dissolution studies in acidic media representative of the industrial process, analysis techniques must be developed for the separation and measurement of long-lived radionuclides (^{14}C, ^{129}I, ^{36}Cl) and tritium. A centrifuge has been designed and will be used for separation of the dissolution residues to facilitate clarification and to measure the moisture content and particle size of the fines.

Studies of actinide and iodine irradiation targets will require the development of a system for gas trapping and analysis during dissolution, especially for measurement and quantification of ^{85}Kr.

The development of a bulk density measurement system by hydrostatic weighing is now being examined for characterization of the precipitates and deposits. This system has been tested under inactive conditions and is expected to be used in 2011 to characterize industrial precipitates.

Industrial dissolution residues will be characterized in the C11 line. These potentially pyrophoric residues must be delivered in a transport cask in which the internal cavity is inerted. Studies and modifications of the package currently used to transport spent fuel to C11 (Padirac RD15IIB manufactured by Getinge-La Calhène) will be necessary to comply with this safety requirement.

5. Conclusion

Conversion and modification of C11-C12 hot-cell line were necessary during ten years of high-activity operation to address new safety and security rules or to adapt the line for new programs. The results obtained since hot commissioning of the line and the programs currently underway demonstrate the quality and flexibility this R&D facility. The future programs are ambitious and will require further modifications and design of new equipment specifically suited to high-activity experiments.

KAERI's Hot cell Experience Learned from DUPIC Fuel Fabrication

2010. 9. 7

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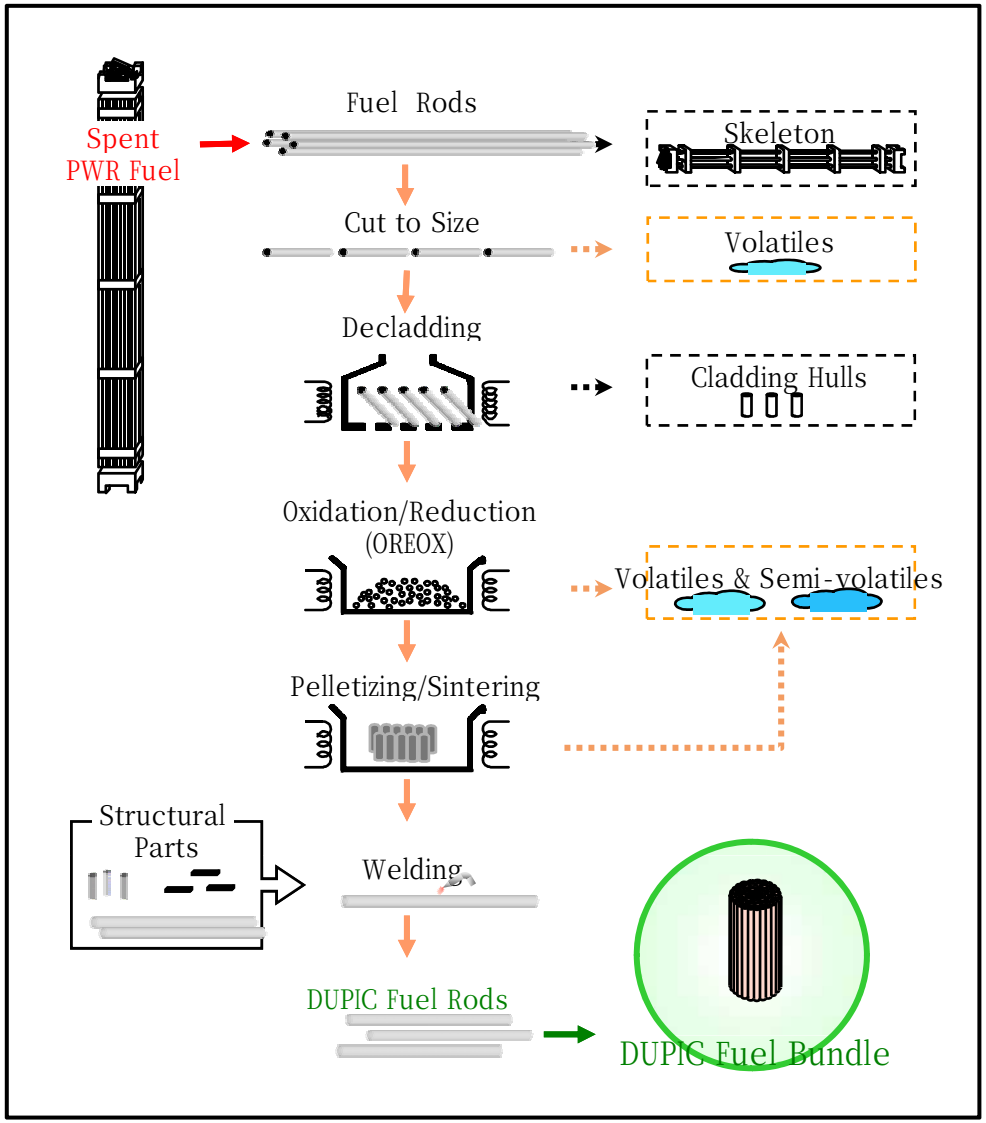
Abstract

- DUPIC (Direct Use of spent PWR fuel In CANDU reactors) nuclear fuel cycle technology that reuses spent PWR (Pressurized Water Reactor) fuel as raw material has been in development at KAERI.
- All DUPIC nuclear fuel fabrication processes should be remotely conducted in a hot-cell because of the high radioactivity of spent PWR fuel.
- Various types of equipment specially designed for DUPIC fuel fabrication have been operated and maintained for 10 years.
- This paper presents the in-cell arrangements of all the DUPIC equipment in the hot-cell and their remote operation and maintenance in situ.

35MWd/kgU

- **U-235 : ~ 0.9%**
- **Pu-239, 241 : ~ 0.6%**
- **FPs : 3 ~4%**

DUPIC Fuel Fabrication



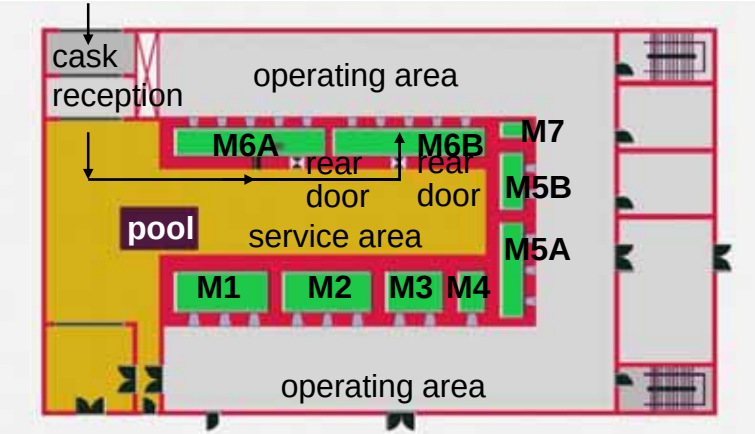
<SF Rodcut>

<Fuel meat>

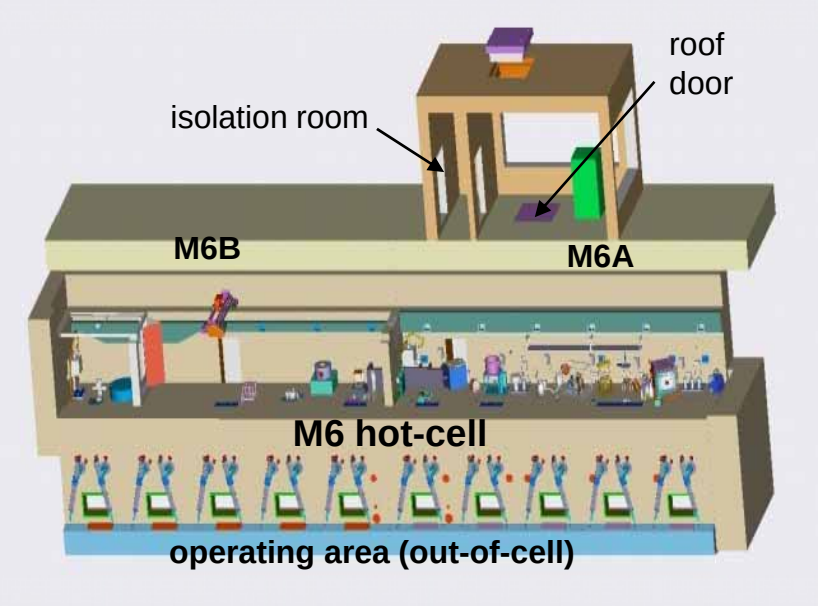
<Off-gas Treatment>

<Remote operation>

DUPIC Fuel Fabrication Facility - *IMEF*



< Layout of M6 Hot-Cell of IMEF >

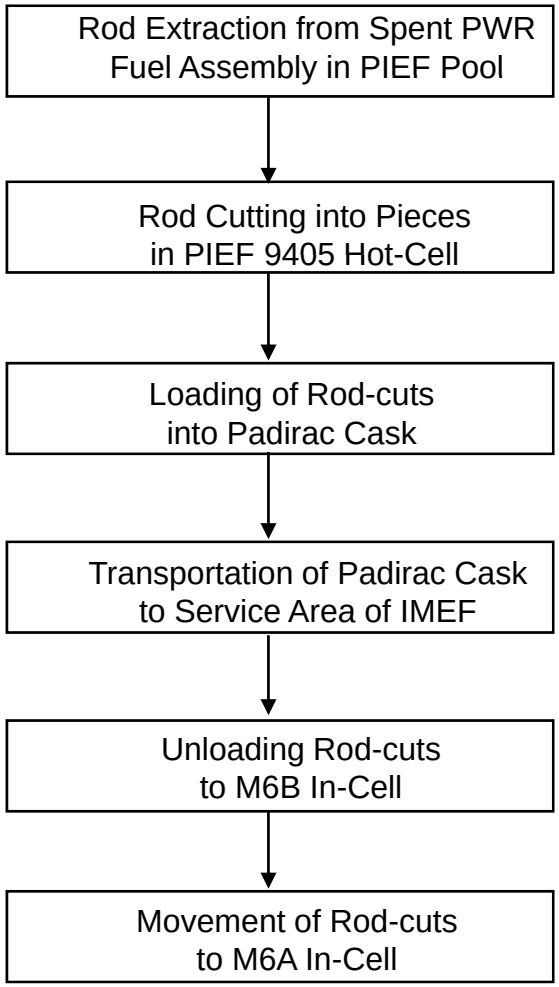


< Graphic Model of DFDF >

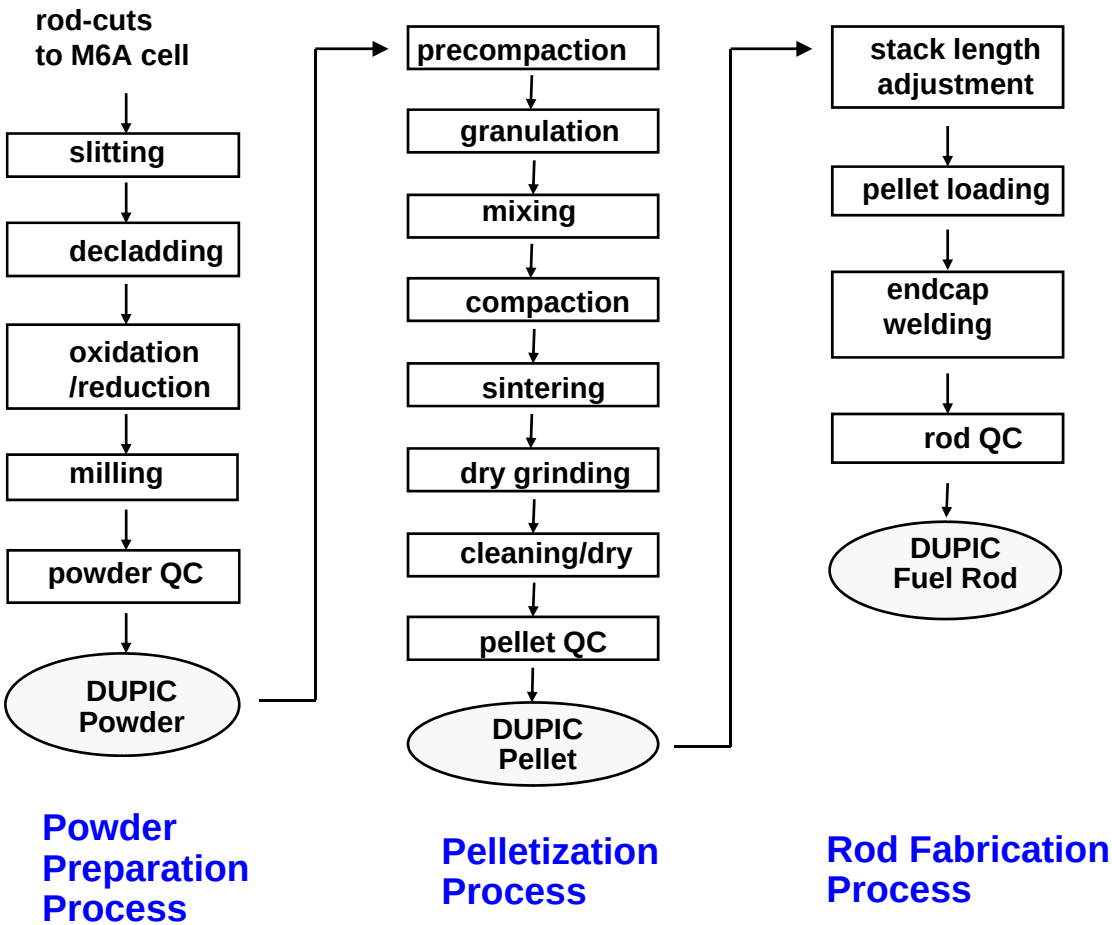
- IMEF (Irradiated Material Examination Facility).
- DFDF(DUPIC Fuel Development Facility)
  - Divided into two rooms – M6A and M6B cells
  - Configuration of each cell: 10x2x4 (LxWxH) m
  - 10 shielding windows
  - A pair of master-slave manipulator on each window
  - Utilities of water, air, gas and electricity provided from the out-of-cell to the in-cell via penetrations
  - 1 overhead crane traversing the length of two cells
  - Access to Isolation room via roof door in M6A cell

DUPIC Fuel Fabrication – *Processes*

❑ Spent PWR Fuel Preparation

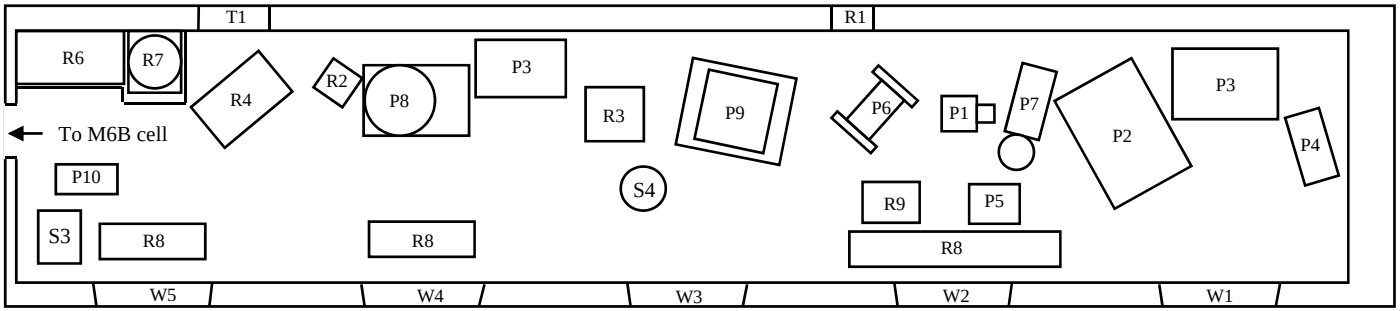


❑ Fabrication Processes

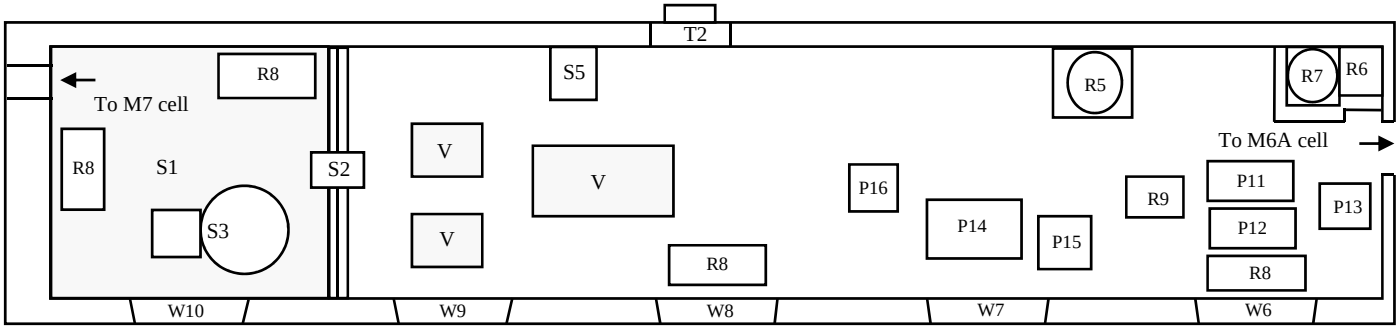


DUPIC Fuel Fabrication – *Equipment Arrangement*

- **Accessibility**
- **Availability of utilities**
- **Visibility**
- **Spatial confinements**
- **Fabrication processes and procedures**
- **Equipment's geometric conditions**



a) Equipment arrangements in the M6A cell



b) Equipment arrangements in the M6B cell

P1: Slitter
P2: OREOX furnace
P3: Off-gas system
P4: Mill
P5: Powder QC
P6: Compaction press
T1: Rear door

P7: Mixer
P8: Sintering furnace
P9: Centerless grinder
P10: Pellet cleaner/dryer
P11: Pellet QC equipment
P12: Pellet adjuster
T2: Rear door with Padirac cask adapter

P13: Pellet loader
P14: Endcap welder
P15: Rod QC equipment
P16: Helium leak tester
R1: Shield plug
R2: Vacuum pump

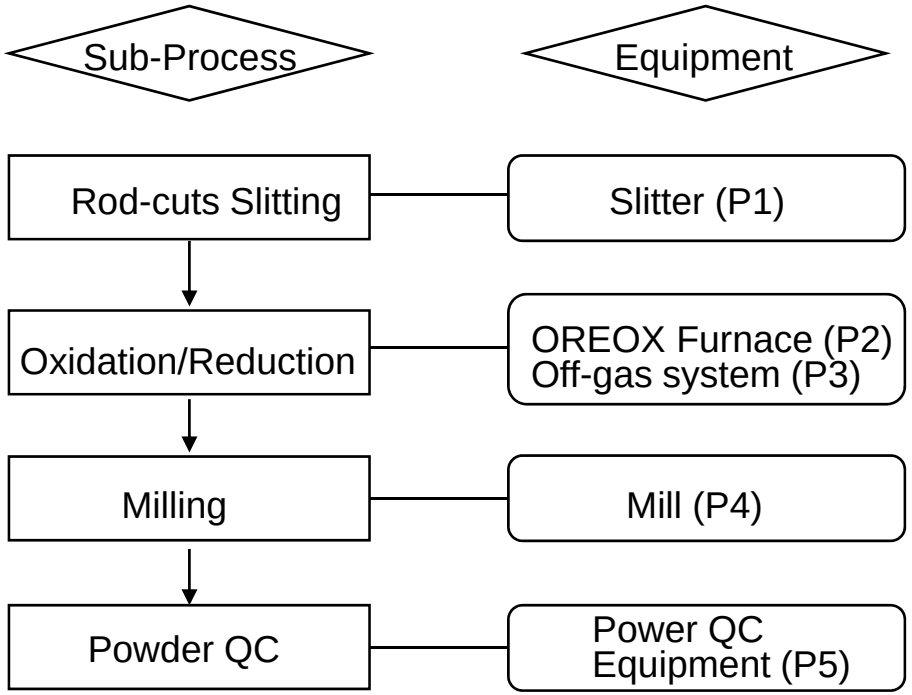
R3: Cutter
R4: QC furnace
R5: DSNC
R6: Material storage
R7: Waste storage
R8: Ventilation filter
W1.. W10: Window

R9: Balance
S1: Decontamination cell
S2: Double door
S3: Dry ice blasting
S4: Vacuum cleaner
S5: Cleaning robot
V: Bundle assembly

In-cell arrangements

DUPIC Fuel Fabrication – *Remote Operation*

❑ Powder Preparation Process



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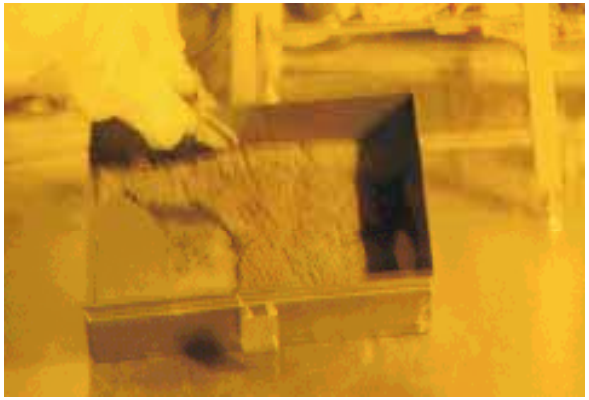
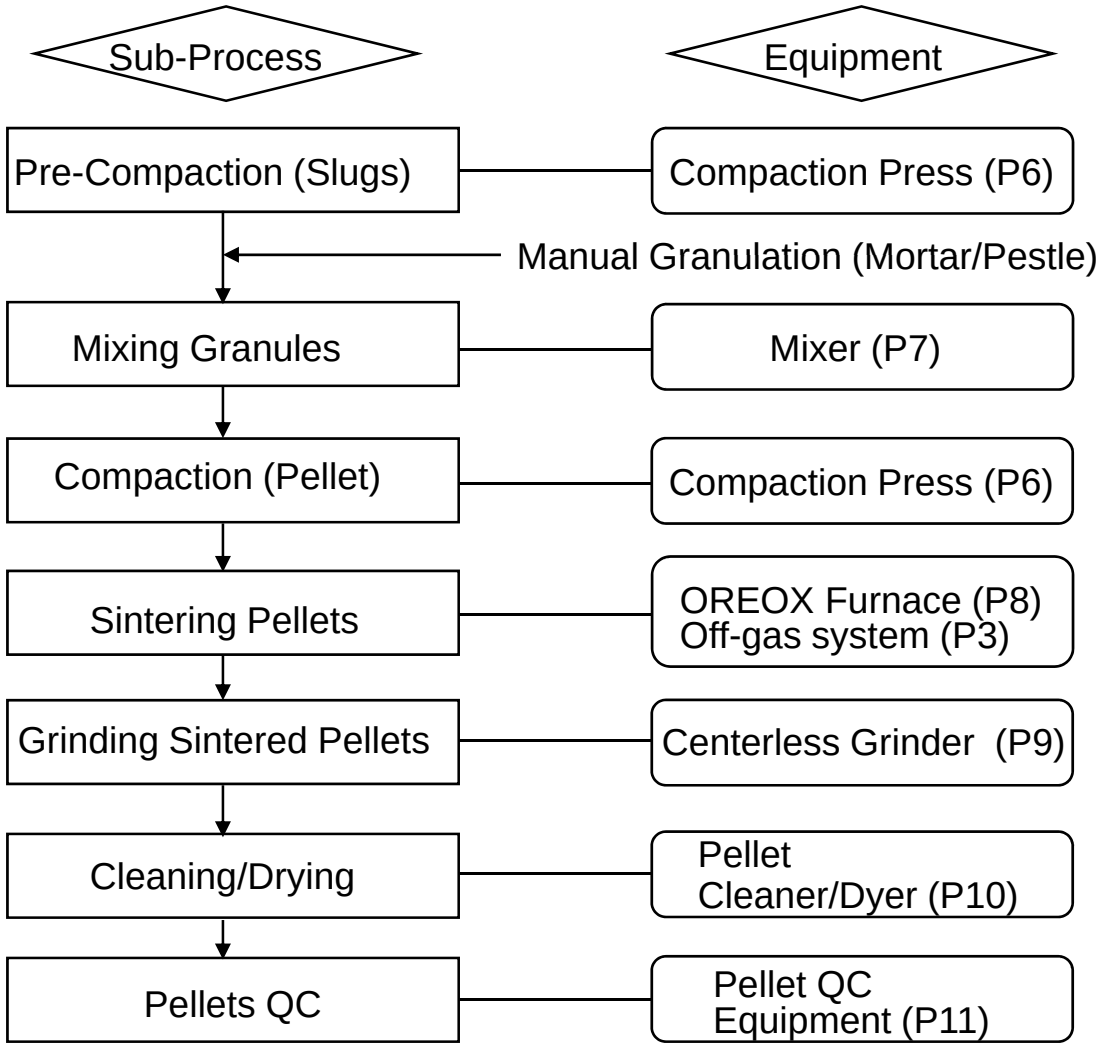
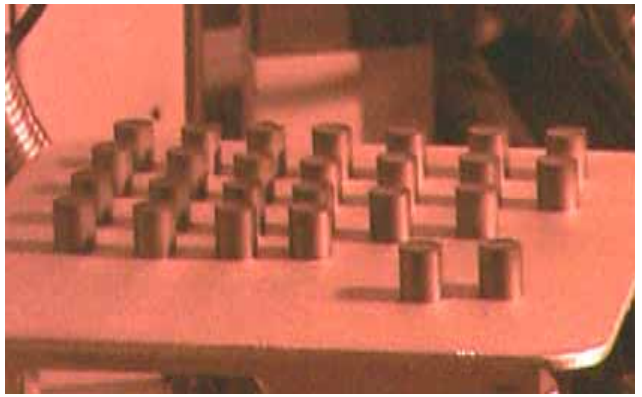


<Remotely Fabricated DUPIC Powder>

DUPIC Fuel Fabrication – *Remote Operation*

❑ Pelletization Process





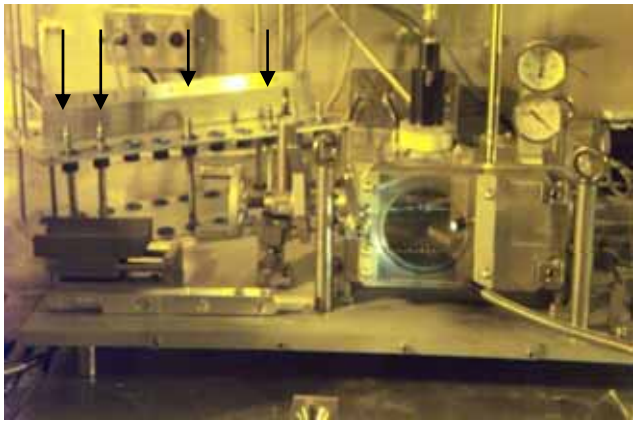
<Remotely Fabricated DUPIC Pellets>

<Off-gas Treatment System>

DUPIC Fuel Fabrication – *Remote Operation*

❑ Rod Fabrication Process

Sub-Process	Equipment
Adjusting Stack Length with Pellets	Pellet Adjuster (P12)
Loading Adjusted Pellets into Empty Rod	Pellet Loader (P13)
Welding End of Pellet-loaded Rod with endcap	Endcap Welder (P14)
Welded Rod QC	Rod QC Equipment (P15)
Examining Leak of Welded Rod	Helium Leak Tester (P16)



<Endcap Welding>

<Remotely Fabricated DUPIC Fuel Rod>

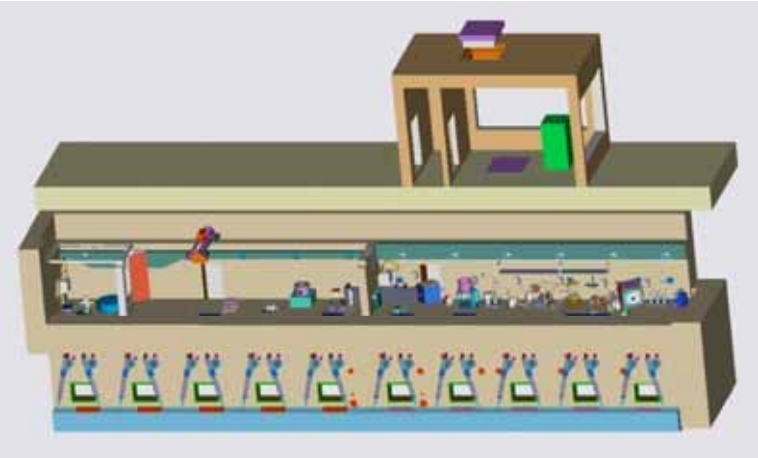
DUPIC Fuel Fabrication – *Remote Operation*

DUPIC Fuel Fabrication – *Remote Operation*

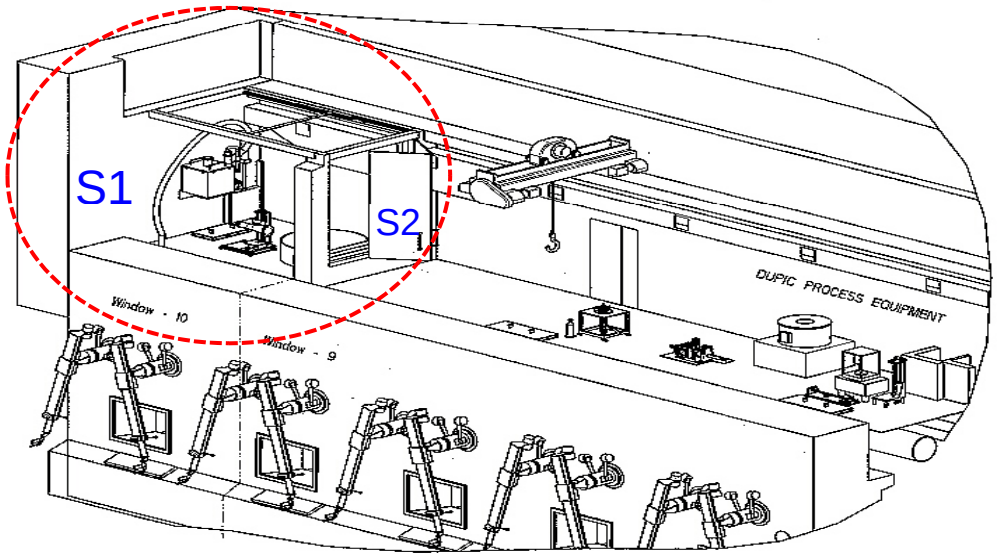
(Movie)

DUPIC Fuel Fabrication – *Remote Maintenance*

- The failed equipment should be repaired on time for normal operation.
- The in-cell maintenance and contact maintenance in the isolation room.
- Decontaminated before maintenance
  - Decontamination cell (S1) is a separate closed room made in the M6B cell : preventing the contamination from spreading during decontamination process
  - Double door (S2) is used to completely close the side of the tamination cell (S1).



<Graphic of the M6 hot-cell and Isolation room>



<Schematic of Decontamination cell>

DUPIC Fuel Fabrication – *Remote Maintenance*

FEDS (Fabrication Equipment Decontamination System)

Consists of an in-cell chamber and a dry ice blasting system in out-of-cell

- In-Cell Chamber
  - a separate closed room constructed inside hot-cell
  - preventing the contamination from spreading inside the hot-cell during a decontamination operation.
- Dry Ice Blasting System
  - utilizes dry ice pellets (solid carbon dioxide) as a blasting media to remove the contaminants deposited on the equipment
  - does not produce second contaminants after a decontamination.

blasting body,

dry ice/air feeding system

blasting/suction part

DUPIC Fuel Fabrication – *Remote Maintenance*

ROCCS (Remotely Operated Contamination Collection System)

- clean the floor surface of hot-cell and collect loose spent fuel debris and radioactive waste.
  - Modular construction of mobile platform, rotary brush tool and vacuum unit for an easy maintenance
  - Small and compact configuration of 300x400x400 (WxLxH) mm
  - Ability to collect loose spent fuel debris and radioactive waste up to 0.3μm by means of the vacuum cleaning method
  - Ability to climb up such obstacles as electric cables, gas and water tubes
  - Small turning radius, and forward, reverse and steering movements

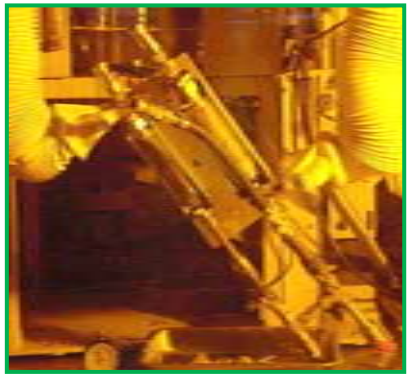
ROCCS-1

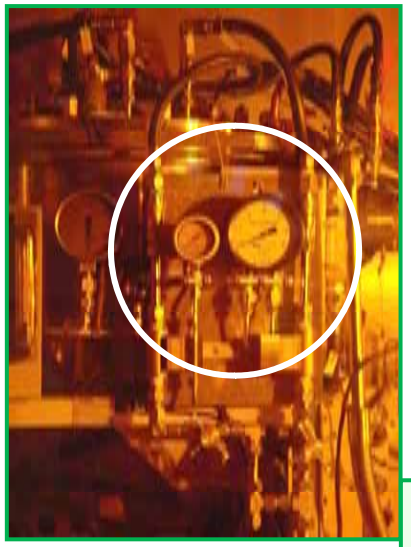
Control Console of ROCCS

ROCCS-II

❖ ROCCS-II : more intelligence in terms of an autonomous navigation, omindirectional cleanup and added vision information

DUPIC Fuel Fabrication – *Remote Maintenance*

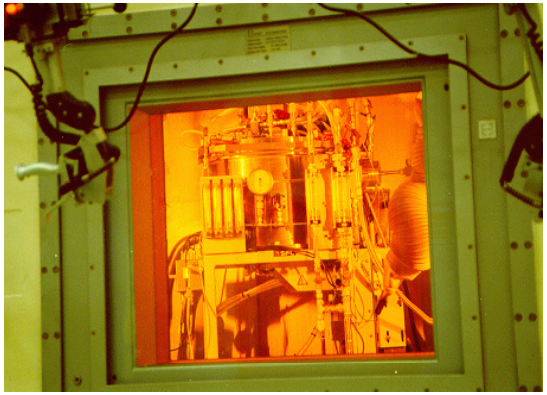


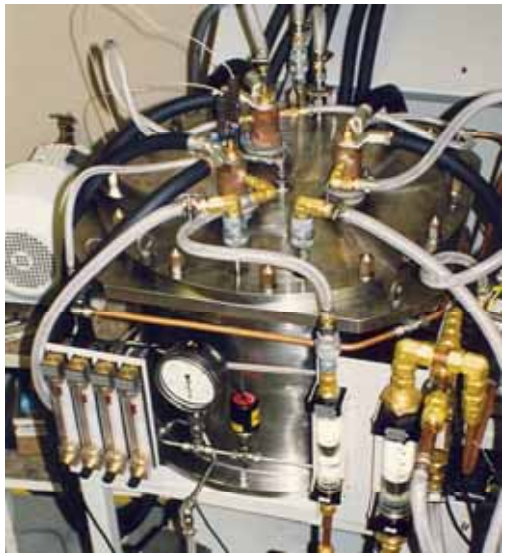


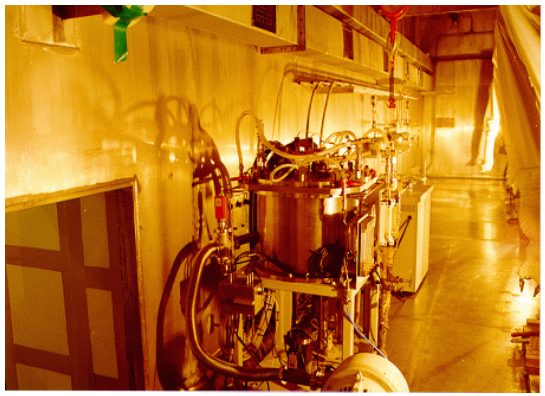
Equipment	Main features of equipment repair
Slitter	• Manufacture slitter for fuel rods with different dimension
Millier	• Improve Jar separation and hopper connection
OREOX furnace	• Flow meter change and improve continuous operation
Press	• Adjust press location level, Side view mirror for dead zone
Grinder	• Change vacuum cleaner for dust collection
Sintering Furnace	• **Change coolant flow system due to radiation damage of PE supportor**
	• **Change vacuum gauge**
	• **Manufacture base plate for loading green pellets**

DUPIC Fuel Fabrication – *Remote Maintenance*

Sintering Furnace
(Heating Element Exchange)







DUPIC Fuel Fabrication – *Remote Maintenance*

Sintering Furnace – *Remote Maintenance*

(Movie)

Summary

- Arrangement and installation of DUPIC nuclear fuel fabrication equipment and devices were completed in late 1999.
- Remote DUPIC nuclear fuel fabrication has been started since middle of 2000.
- 850 DUPIC pellets, 16 mini-elements and 6 DUPIC fuel rods have been fabricated until now
- 5 mini-elements have been irradiated in the HANARO research reactor at KAERI
- DUPIC fuel bundle fabrication technology is under development.

HOT CELL DESIGN CONSIDERATIONS FOR MEDICAL ISOTOPE PRODUCTION FACILITIES

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ABSTRACT

With a limited list of producers and a growing demand, there is an increased interest in developing additional medical isotope production capability worldwide. This situation has been exacerbated by aging target irradiation reactors and production facilities ceasing operations in Canada and the Netherlands in early 2009.

This paper presents specific planning, design, and construction considerations associated with post-irradiation target processing relating to handling Low Enriched Uranium (LEU) targets, using acid vs. alkaline processing, handling process gas, and operating the associated waste processing steps. Additional aspects include cell arrangement and layout, vertical vs. horizontal arrangement, process equipment selection, window selection, manipulator selection, ventilation equipment, remote operations design, and remote or hot maintenance.

Introduction

Recent or pending global loss of production capacity for Mo-99 has sparked interest on the part of many reactor and accelerator operators, who are exploring opportunities to use their irradiation capabilities for Mo-99 production. However, most of the irradiation facilities being considered, particularly those facilities not principally focused on nuclear science or engineering, lack the hot cells essential for post irradiation mechanical and radiochemical separations that ultimately produce medically usable Mo-99.

The objective of this paper is to identify the hot cell design considerations associated with medical isotope production facilities. The discussion is focused on those facilities that produce Mo-99 sources that, in turn, are used as sources for Tc-99m in various medical applications.

The discussions in this paper are from the perspective of the mechanical engineering designers of hot cells and the associated operating systems, such as windows, manipulators, transfer equipment, process piping and valving, and so forth. As mechanical designers, we work closely and iteratively with the chemical engineering process designers. This paper does not discuss the relative merits of acid versus alkaline dissolution. We focus on designing the “machine” – the remotely operable and maintainable equipment, the “goes into” and “goes out of” mechanisms, etc.

Top Level Design Objectives

As always, the top level design objective for hot cells is to provide and maintain confinement and shielding. In the case of medical isotope production, we are dealing with nuclides with short half lives, so the hot cells should not be on the process critical path between the end of target irradiation and the shipment of medically usable products. Our additional top level design objective is that the material-in-process should never be sitting idly (and decaying away), while some hot cell function is being performed.

Some specific approaches to minimize processing time include:

- Capability to receive and process fully loaded transfer casks from target irradiation while maintaining criticality safety
- Target design features to ease disassembly
- Special tooling and fixtures for facilitating mechanical disassembly
- Maximizing speed of dissolution step with vessel design and maximum exposure of targets to solvent
- Multiple separation lines for redundancy as well as throughput
- Separation of waste management steps from production steps
- Automation for downstream packaging and labeling steps

Automation is not recommended for the early steps due to the high radiation fields associated with the freshly irradiated targets and the difficulty in localized shielding for the control mechanisms. However, at the end of the process line automation can be used in the packaging steps as the radiation fields are much lower and the equipment is not exposed to damaging radiation.

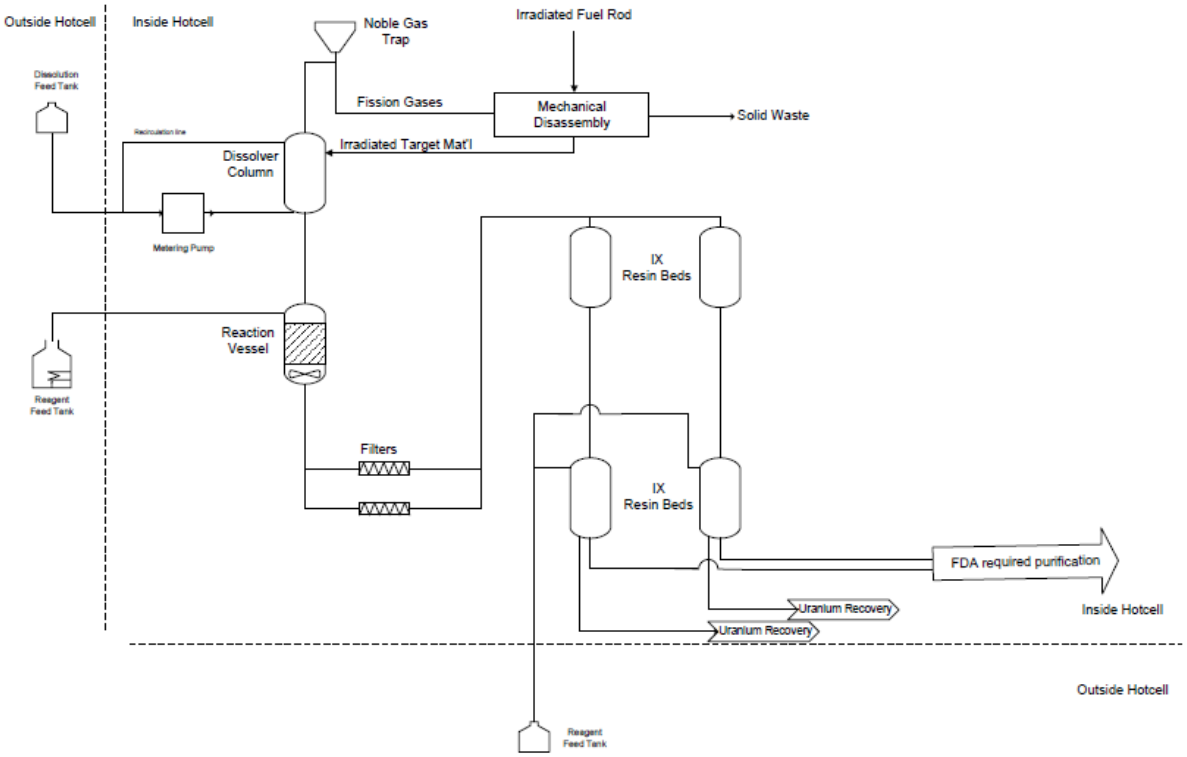
PROCESS ZONES

Regardless of the initial target configuration and irradiation options, whether the Mo-99 is produced as an activation product or fission product, whether in a reactor or in an accelerator, whether from HEU or LEU, whether from metal or oxide target material, there are the same general process steps that must be accomplished within (or in close connection with) hot cells.

- Receipt
- Mechanical disassembly
- Dissolution
- Filtration & radiochemical separation
- Product purification, packaging, transportation
- Waste management
- Maintenance

It is not necessary or cost effective to have a separate hot cell for each one of these activities, but the overall hot cell facility should be designed to provide a dedicated and independent workstation for each activity. A “workstation” is meant to be an in-cell location within the view and operational reach and control of a single operator – classically, this means a shielded window with a pair of through-the-wall master-slave manipulators, an in-cell work surface with appropriate tooling, lighting, HVAC, and capability to transfer materials to-and-from. “Independent” means that the workstation’s functions can be accomplished without interferences with adjacent workstations or out-of-cell activities, and without unwanted cross contamination with other workstations.

PROCESS BLOCK FLOW DIAGRAM



NEW VERSUS RETROFIT

It is the authors' experience that retrofitting an existing hot cell facility, originally designed for different functions, takes as long (or longer), costs as much (or more), and ends up with more design compromises and operations inefficiencies than a Greenfield facility. However, there are frequently other considerations (e.g., space availability) that necessitate retrofitting of an existing hot cell facility. One of the most significant obstacles to retrofitting existing facilities is the necessity to minimize processing time and the requirements for having all the steps adjacent to each other and minimizing time of transit from one step to the next. Additionally, the early steps of receipt, mechanical disassembly, and dissolution require significantly more shielding than the later steps, and many existing facilities are either under-shielded for the early steps or over-shielded for the latter.

GENERALIZED ARRANGEMENTS

There are several approaches to arrangements of cells for hot processing, including:

- Adjacent but separate cells
- Traditional cells with horizontal transfers
- Traditional cells with horizontal transfers connected to a common isolation room
- Great Room, with isolation boxes
- Vertical cells
- Modular cells

Of these types the authors believe that traditional cells with horizontal transfers connected to a common isolation room offers the most advantageous arrangement for isotope production. This arrangement provides the ability to easily move materials from one processing step to another, minimizes cross contamination between process steps, allows for selective and appropriate shielding for each step, decouples waste management and maintenance activities from the process steps, and provides space for handling and maintaining the process equipment.

Vertical arrangements have also been proposed to achieve some of these same advantages; for example, by placing waste management activities in cells below the processing steps, allowing gravity to do some of the work of moving solutions and also allowing decoupling of the process steps from the waste management steps. Adjacent but separate cells and modular cells can be used for isotope production, but the need for packaging, shielding, and contamination control for the movement of materials from one cell to another significantly increases the production time and reduces the activity of the end product.

DECOUPLE ANCILLARY FUNCTIONS

One design objective essential for the creation of an efficient medial isotope production hot cell facility is to decouple all of the ancillary functions not on the process critical path. The primary "ancillary functions" that must be performed elsewhere (and else-when) from production are: waste management, manipulator repair, reagent preparation, transport flask (cask) cleaning, repair, and staging, and general housekeeping. Decoupling can be accomplished by arrangement and location of the process lines and waste management lines, by location and tooling for maintenance activities, and by appropriate placement of ports for moving material in and out of the cells. Decoupling is important to facilitate minimum processing time and maximum activity in the end product.

WINDOWS, OBSERVATION PORTS, CAMERAS

Viewing of operations is necessary, especially when equipment replacement or adjustment is required. The most common hot cell design includes the use of leaded glass shield windows, which can be sized and installed to give tremendously wide ranges of view throughout the cell (although considerable distortion may occur at extreme viewing angles). Windows have long been the default choice for viewing the cell interior, at least within the US DOE complex. However, windows are very expensive, for the materials, the design, the installation, and continuing maintenance. [The authors got a recent budgetary quote for windows with 3.23 gm/cm3 glass, without non-browning additives, installed, for about $US 12/ cubic inch (approximately € 0.56/ cm3).]

One alternative for consideration in isotope production hot cells is the use of observation ports rather than windows. These are small diameter, leaded glass filled tubes that penetrate the shield wall. Unlike windows, observation ports are set to give direct views of only one relatively small area, such as an ion exchange column mounting and connections. The principal advantage of observation ports versus windows is initial cost savings because much smaller volumes of leaded glass are required.

A final alternative for direct viewing is a stand-alone camera system. [Windows and observation ports are typically supplemented by cameras for viewing areas outside of direct viewing range, so camera systems are installed in virtually all hot cells, anyway.]

HOT REPAIR AND SUPPORT AREAS

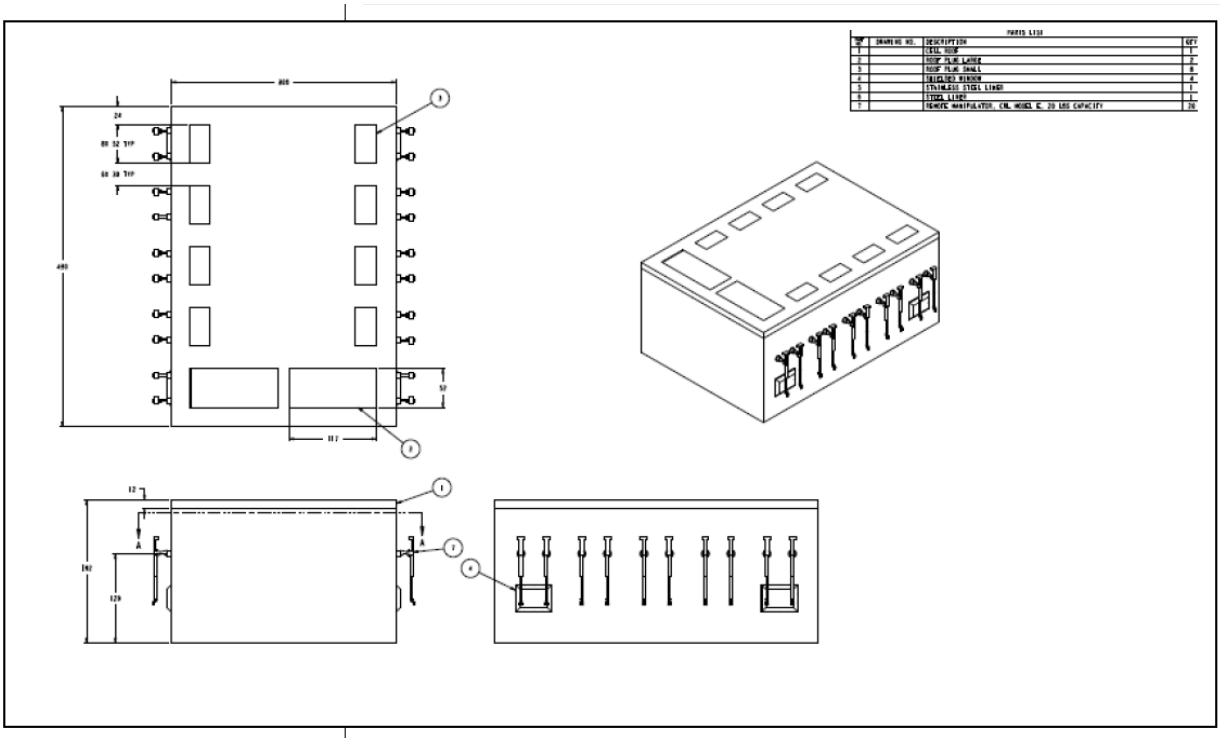
As with all shielded operations it is important to identify and provide for space for the hot repair of process systems and equipment, as well as manipulators and other mechanical equipment associated with the facility (utilities, HVAC, etc). Some specific design considerations that should be applied for process maintenance include:

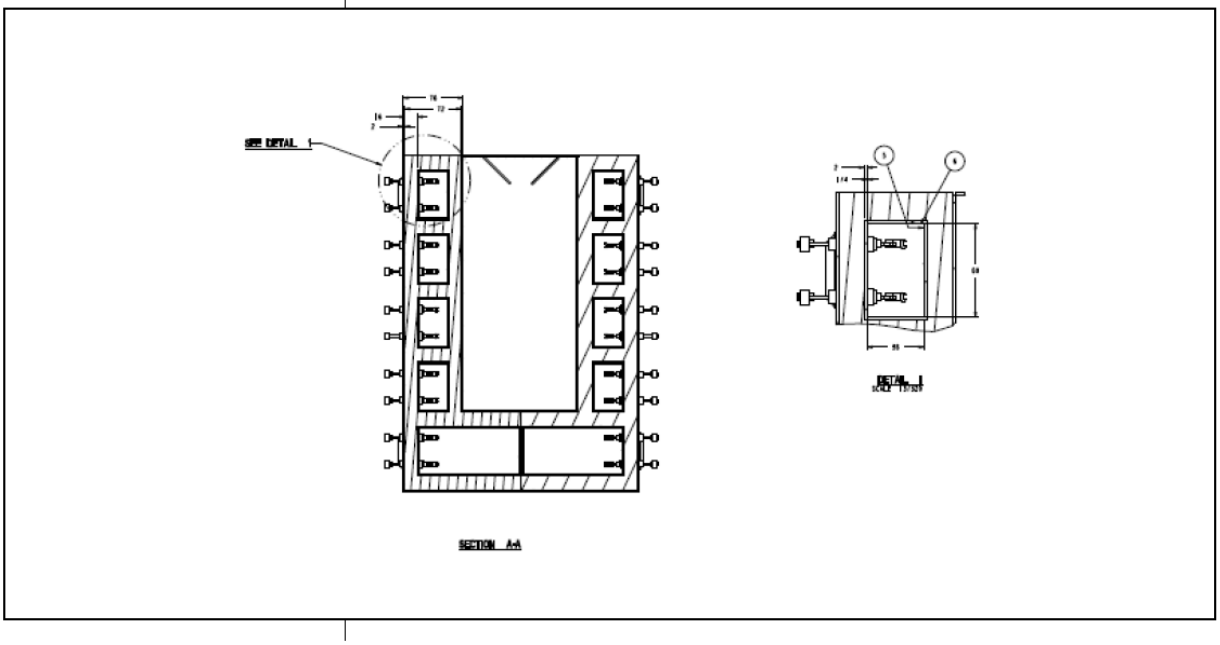
- Enough space to pull manipulators for maintenance and/or replacement
- Ability to recover and repair any overhead cranes
- Design of process equipment for long life and/or easy of removal and replacement
- Sizing of equipment to facilitate removal and replacement
- Design of process equipment to minimize hold-up and creation of "hot zones" within the equipment
- Sizing of ports, plugs, and penetrations to allow for movement of materials and equipment in and out of the cells

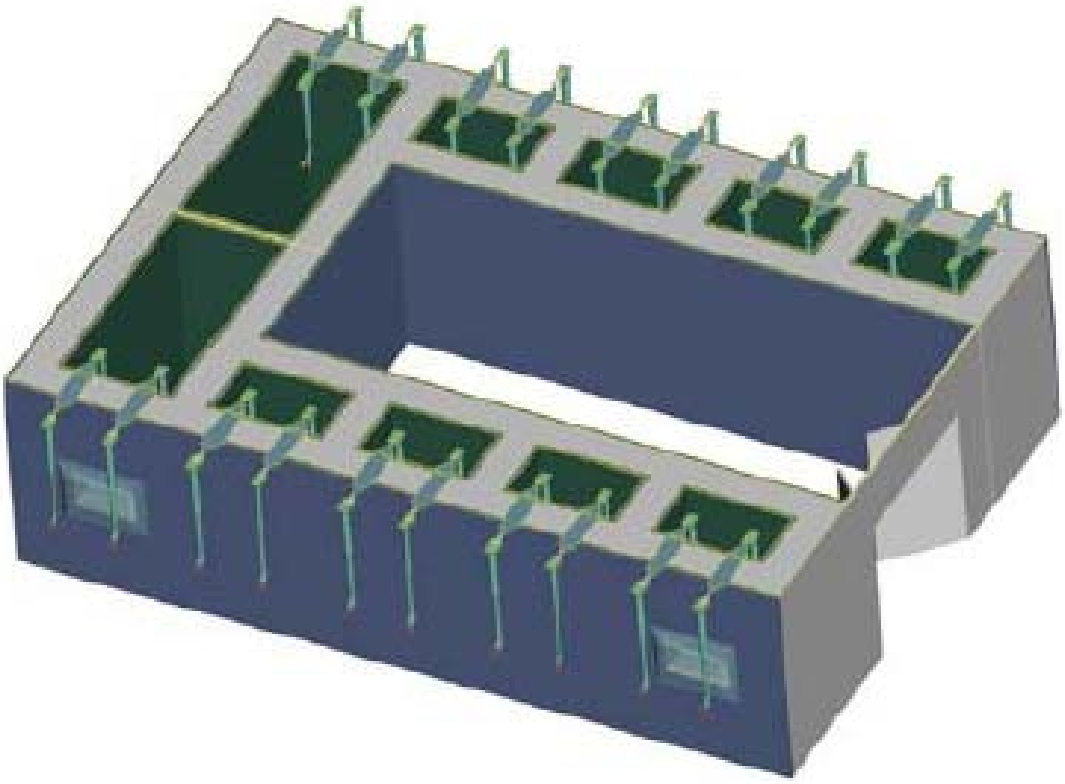
- Sizing of manipulators and cranes to handle the loads from equipment and systems

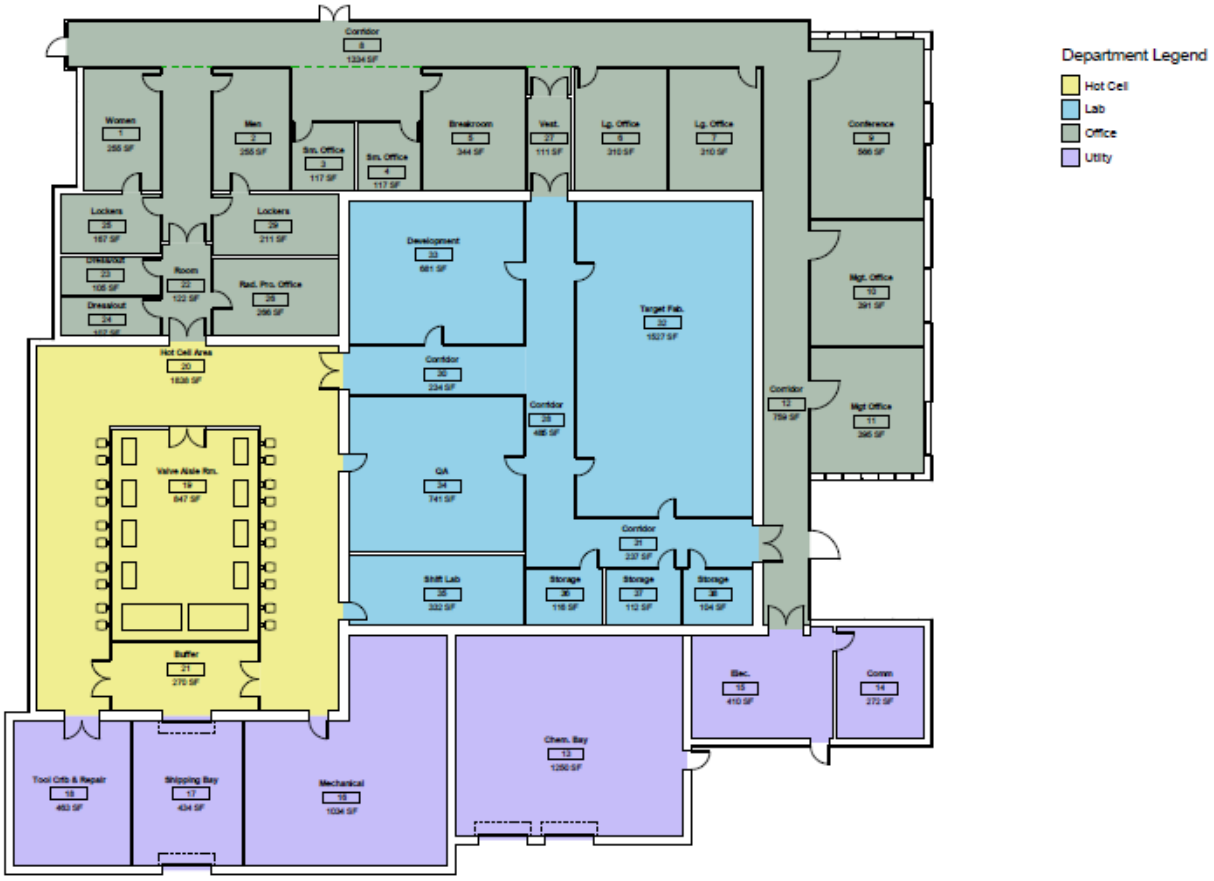
Critical to the design of the cells is the strategy for maintaining the process and material handling equipment. Incorporation of redundant equipment and the operational flexibility to ensure that in the event of an equipment failure, that material in process can be quickly transferred to allow further processing while the malfunctioning equipment can be addressed independently is crucial. Once the overarching cell operating and maintenance philosophy is developed, the requirements for remote operating capabilities, adjoining repair areas, and any special equipment that may be required to execute the strategy may be developed.

PUT IT TOGETHER – ONE POSSIBLE CONCEPT








Corridor
8
1334 SF
Women
1
255 SF
Men
2
255 SF
Srs. Office
3
117 SF
Srs. Office
4
117 SF
Breakroom
5
344 SF
Vest.
27
111 SF
Lg. Office
6
310 SF
Lg. Office
7
310 SF
Conference
9
566 SF
Lockers
25
167 SF
Lockers
29
211 SF
Dress/Undr
23
105 SF
Room
22
122 SF
Rad. Pro. Office
28
286 SF
Dressout
24
Development
33
661 SF
Target Fab.
32
1827 SF
Mgt. Office
10
391 SF
Hot Cell Area
20
1838 SF
Corridor
30
234 SF
Corridor
29
486 SF
Corridor
12
754 SF
Mgt Office
11
395 SF
Valve Aisle Rm.
19
647 SF
QA
34
741 SF
Corridor
31
237 SF
Buffer
21
270 SF
Shift Lab
35
332 SF
Storage
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116 SF
Storage
37
112 SF
Storage
38
104 SF
Elec.
15
410 SF
Comm
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272 SF
Chem. Bay
13
1260 SF
Tool Crib & Repair
16
463 SF
Shipping Bay
17
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Mechanical
18
1034 SF
Department Legend
Hot Cell
Lab
Office
Utility