MANAGING PAST AND DAMAGED SPENT FUEL

(Session 9)

Chairperson

E. DeLEON
USA

MANAGEMENT OF DAMAGED SNF HANDLING OPERATIONS AT PAKS NPP

E.A. ZVIR[*], V.P. SMIRNOV[*], I. HAMVAS[**]
[*] Research and Development Company "Sosny"
Moscow
Russian Federation
[**] Paks NPP
Paks
Hungary

Abstract

The issue of handling leaky fuel is one of the crucial issues of nuclear energy. It is directly connected with transportation of leaky spent nuclear fuel (SNF) that is mostly stored in the station's cooling pools. At present damaged spent nuclear fuel (SNF) of thirty VVER-440 spent fuel assemblies (SFA) is loaded into the ventilated canisters of types 28 and 29 and temporary stored in the cooling pool of Paks NPP. The report presents the milestones of preparation and safety justification of the technology for preparing canisters with damaged SNF of Paks NPP for transport to FSUE "PA "Mayak" (Russia) for reprocessing.

1. INTRODUCTION

On 10 April 2003, during the outage period a chemical cleaning program for the fuel assemblies was carried out at the unit 2, in a specially designed cleaning tank. The tank was located in a pit no. 1 near to the reactor. 30 fuel assemblies were significantly damaged due to inadequate cooling.

In the period of 01.10.2006–30.03.2007, the NPP and Russian specialists succeeded in arranging and performing activities to remove the content of the cleaning tank. All SNF (more than 5 tons) from the cleaning tank was loaded into the specially desined ventilated canisters of types 28 and 29 and it is temporary stored in the spent fuel cooling pool of Paks NPP.

Storage of SNF in the cooling pool in ventilated canisters filled with water requires development of the basic technology for subsequent handling.

Taking into account the state of the assemblies, the optimum solution would be to reprocess the spent fuel and dispose of the resulting waste.

In order to justify safety of management of damaged spent fuel, on behalf of Paks NPP Russian specialists have performed a significant amount of scientific research and design development work. The results became the basis for *the basic technology* for preparation of the canisters containing damaged spent fuel for transportation to Mayak for re-processing.

2. PREPARATIONS

The overall process of canisters management for transportation, shipment from Paks NPP to Mayak and further treatment provides for the following three key stages (shown on Figure 1):

- Handling of the canisters at Paks (preparation for shipment; removal of the canisters from the cooling pools);
- Transport of the canisters from Paks to Mayak;
- Re-processing of the fuel at Mayak.

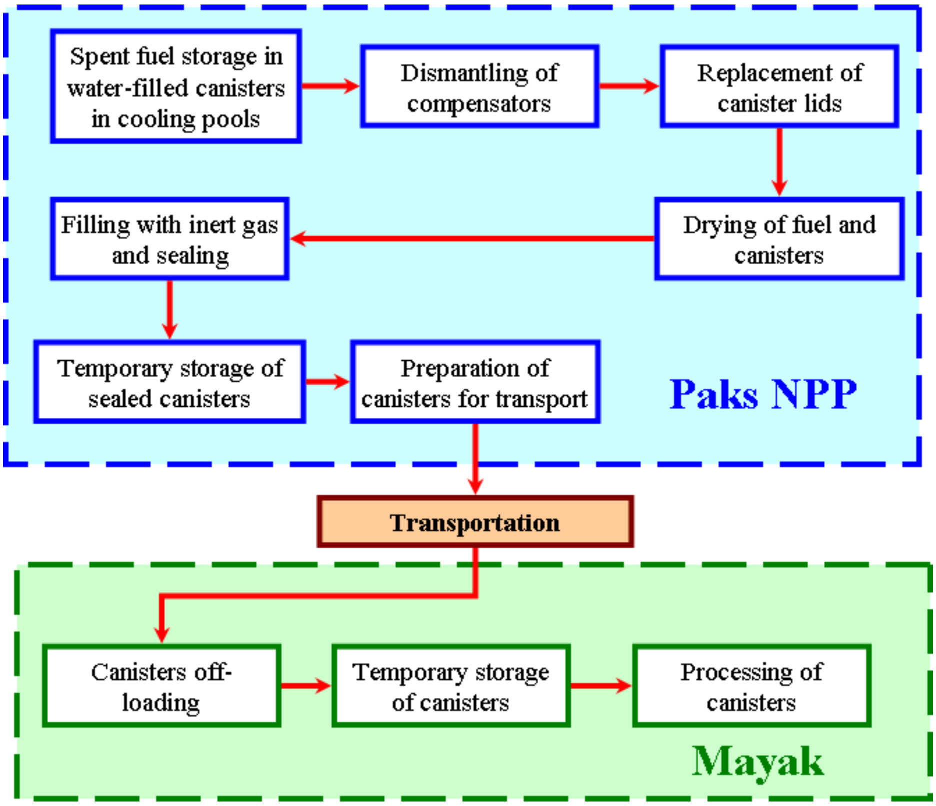

FIG. 1. Generalized schematic of canisters management.

The main difficulty with the assurance and justification of safety of transportation of wet damaged spent fuel is the need to keep the canister leak-tight over an extended period of time (to one year). This duration consists of the time the leak-tight canisters would be kept at the plant site, the time to deliver and receive to Mayak and the time the fuel will stay at Mayak before being actually cut up to be re-processed.

Transportation of wet damaged SNF differs from traditional way of handling intact SNF first of all due to presence of radiolysis of the residual water in the canisters. Radiolysis results in increase of pressure inside tight canisters and generation of explosive gaseous mixture. Based on the experience in handling damaged fuel at TMI-2 NPP it is one of the main processes influencing safety [1].

During the development of the basic technology, analysis was carried out of the Russian and international regulations, taking into account the safety requirements for management of damaged SFAs at the re-processing plant, and a justification was produced for transportation of failed spent fuel from the Paks site to Mayak. All key safety aspects of transport were considered: nuclear and radiation safety, structural integrity and leak-tightness of the canisters, thermal conditions, explosion and fire safety.

The preliminary calculations performed by the Russian Institute of Experimental Physics (VNIIEF) confirmed that defective spent fuel can be safely transported in gas-filled canisters types 28 and 29, the design of which takes into account the requirements of Mayak to handling prior to re-processing.

For the purposes of transportation, the transport overpack TUK-6 could be used with basket 13 (for type 28 canisters) and basket 12 (for 29 canisters). For loading of type 28 canisters into basket 13, additional transport inserts would have to be designed and fabricated. So that all canisters can be transported from Paks to Mayak, just four TUK-6 overpacks would be sufficient.

3. CHOICE OF CANISTER DRYING METHOD

Residual amount of water in the canisters must not exceed 5% of the mass of fuel; this requirement is attributable to the specifics of spent fuel re-processing at Mayak. Accordingly, water must be removed from the canisters before they are shipped to Mayak.

Preparation of canisters at Paks for transportation to reprocessing may be performed following these processes:

- Water removal with no additional drying, filling with an inert gas and sealing of the canisters;
- Water removal, drying of spent fuel, filling with an inert gas and sealing of the canisters;
- Drying of spent fuel without prior water drainage, filling with an inert gas and sealing of the canisters.

In order to select the optimum technology for preparation of the canisters for shipment, experimental equipment was designed and fabricated, and tests carried out to try the three approaches to water removal — on reduced-height simulators of type 29 canisters, loaded with irradiated VVER-440 fuel, and on full-scale type 28 and 29 canisters loaded with dummy spent fuel assemblies.

Electrotechnical casting porcelain (ECP) was selected to simulate damaged SNF. Thermophysical parameters of electrotechnical casting porcelain, UO_2 and ZrO_2, being important for drying, are provided in Table 1.

Heat capacity of the volume unit of ECP and UO_2 are close in values, i.e. the amount of heat required to heat ECP and UO_2, and consequently heating time are comparable. Thermal conductivity of ECP is 3 time less than thermal conductivity of irradiated UO_2 that gives an opportunity to simulate conservatively UO_2 drying using ECP. Simulators of fuel pellets were fabricated from ECP (Fig. 2).

TABLE 1. THE BASIC THERMOPHYSICAL PARAMETERS OF MATERIALS

No	Material	Density at 10% porosity, g/cm^3	Thermal conductivity at 0°C, W/m°C	Heat capacity at 20°C, G/kg°C	Heat capacity in 1 ml of volume, G/°C
1	UO_2 irradiated up to 30 MW·d/kg U	9,7	6	230	2.2
2	ZrO_2	5.2	2	502	2.6
3	ECP (SiO2~65%, Al_2O_3~24%, Fe_2O~0.6, TiO_2-0,5%, CaO-0.4%, Mg~0.3%, K_2O+Na_2O~3.5%, K_2O/Na_2O = 2-3.3)	2.3	2	900	2.1

FIG. 2. Simulators of fuel pellets from electro-technical casting porcelain.

Density of ECP is lower than the density of UO_2 and ZrO_2, i.e. particles of ECP of the same size are lighter than particles of ZrO_2 and UO_2. Therefore there will be more small particles carried out by the steam flow from the canister in the process of drying ECP than in the process of drying the full-scale canisters of types 28 and 29.

ECP was used also for simulating damaged ZrO_2, since thermal conductivity of ECP and ZrO_2 coincide and difference in heat capacity of the volume unit of these materials is not more than 20%. For conservative simulation of drying of “spillage” consisting of damaged pellets of UO_2 and oxidized fuel claddings, pellets from ECP were milled to the particle size less than 2,5 mm (Fig. 3).

To obtain grinded ZrO_2 standard unirradiated claddings of VVER-440 fuel rods fabricated from Zr-1%Nb alloy were oxidized in a muffle furnace at the temperature of 1000°C. After that they were crumbled as shown at Fig. 4.

FIG. 3. ECP with the particle size less than 2.5 mm.

FIG. 4. Grinded ZrO_2.

To determine the impact of the structural material oxidation extent on the drying speed simulators of cladding fragments with the bottom plugs of 300 mm long (Fig. 5) were fabricated from standard unirradiated claddings of VVER-440 fuel rods, half of the simulators was oxidized in a muffle furnace at the temperature of 1000°C (Fig. 6). Besides, the trimmed simulators of oxidized and non-oxidized fragments of the fuel rods beam were fabricated (Fig. 7).

FIG. 5. Simulators of non-oxidized fragments of fuel rod claddings.

FIG. 6. Simulators of oxidized fragments of fuel rod claddings.

Two types of SNF simulators used in the experiments for loading full-scale canister 28 are provided in Fig. 8. The first type is the full-scale beam of VVER-440 fuel rods, structural elements of which are not oxidized. Fuel rods represent the claddings (outer diameter of 9.1 mm, inner diameter of 7.72 mm) with the plugs at both ends and there is a throughout defect of 0.5 mm in diameter in the area of the upper plug. The claddings are filled with the simulators of the pellets from ECP (height of 10 mm, outer diameter of 7.6 мм, diameter of the central hole of 1.2 mm), the total height of the fuel column is 2420 mm. The second type is the fuel rods beam fragment of 1650 mm long, the structural elements of which are not oxidized. The fuel rods represent empty claddings with the plug at the bottom and removable part of type 6 of canister 28 of 850 mm high completely filled with particles of ECP of the size less than 2.5 mm.

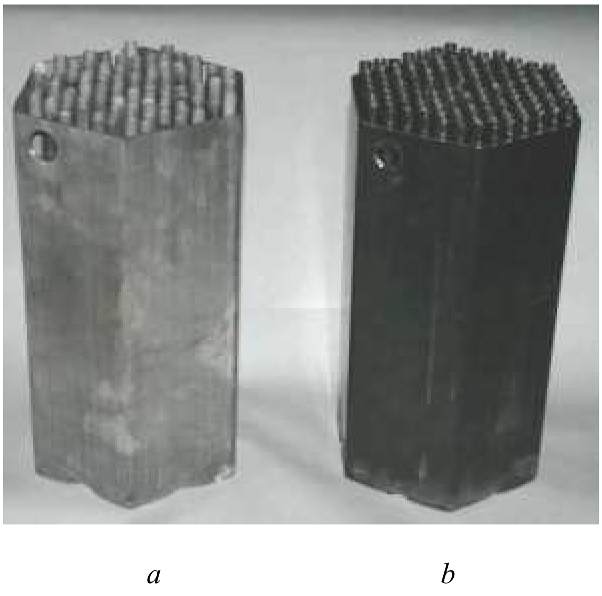

FIG.7. Trimmed simulators of oxidized (a) and non-oxidized (b) fragments of fuel rod beam.

FIG. 8. Simulators of SNF for canister 28.

After removal of water, some moisture will still remain in the canisters, and the process of radiolysis may cause pressure to rise above permissible, and a highly explosive mixture of hydrogen and oxygen may form. In order to determine the amount of residual moisture in the canister that would guarantee safety at all stages of management, experimental research was performed to model the storage of deteriorated spent fuel in a leak-tight canister, after water removal by various methods. Fuel from VVER-440 FA with the following parameters was used for experiments:

- Initial enrichment by U^{235} — 3.6%;
- Average burnup along FA — 27.6 MW·d/kgu;
- Maximum burnup along FA height — 32.0 MW·d/kgu;
- Total FA operation time — 649.1 eff.d;
- Date of discharge of FA from the reactor — 18.06.99.

Logical sequence of experiments is provided in Fig. 9. Level-one experiments determined the conservative with respect to radiolysis condition of spent fuel for all types of canister loading at Paks.

The testing rig to examine behavior of damaged VVER-440 fuel under conditions simulating SNF storage in tight canisters was created to perform level-two experiments.

Level-two experiments measured the rate of generated of hydrogen, oxygen and gaseous fission products into the canister cavity depending on the amount of residual water and ambient temperature. These results are conservative with regard to Paks NPP fuel loaded into canisters of types 28 and 29.

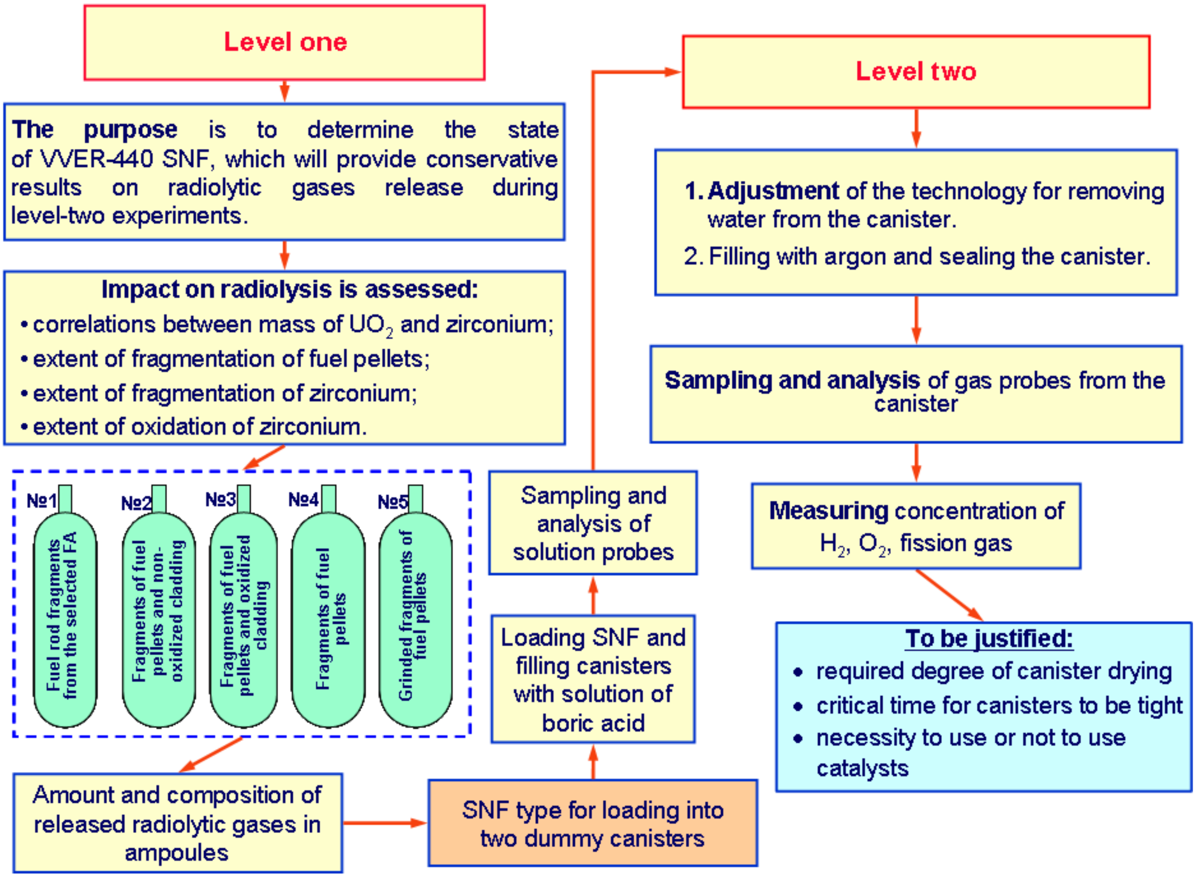

FIG. 9. Logical sequence of experiments.

Based on the results of the experiments the following conclusions were made:

- Justification of safety at all stages of management of canisters sealed after water drainage is impossible, as over six months volumetric concentration of hydrogen in a type 29 canister with free cavity volume 20 litres will reach 8.7%;
- If, however, after the end of thermo vacuum drying the residual pressure in the canister does not exceed 6 mbar, then after two years of storage in a leak-tight type 29 canister with free cavity volume 18 litres volumetric concentration of hydrogen will not be greater than 0.05%, whereas in a 28 canister with free cavity volume 50 litres it will not be higher than 0.04%;
- The amount of hydrogen that the canister would accumulate over twice as long in a leak-tight condition, would be two orders of magnitude below the level of self-ignition for the oxygen-hydrogen mixture;
- With leak-tightness class IV (permissible leakage up to $6.7 \cdot 10^{-7}$ Pa·m^3/s), the time it will take for the excessive pressure in the canister to drop is many times greater than double the duration of time from the beginning of sealing of the first and completion of processing of the last canister;
- Since there is lack of oxygen in the canisters tightened after thermal vacuum drying, it is not efficient to install catalysts in them.

Taking into account the estimated duration and exposure, as well as the quantity of resulting radwaste, thermo vacuum drying without prior drainage was identified as the optimum approach.

Safety of canisters management prepared and sealed following this process and nuclear safety of the process of spent fuel accumulation on the filtering elements were confirmed by the experiments. It was determined that the release of gaseous fission products during drying the canisters would not exceed the reference level established at the Pask NPP. Dilution with air has been proposed for non-condensing gases as they are pumped from the canister to prevent formation of explosive mixes.

As a result of the research and development activities, experimental results and analytical evaluations have been obtained that are needed for licensing of the canister drying technology with no prior drainage of the boric acid solution.

4. BASIC TECHNOLOGY OF CANISTERS PREPARATION

The basic technology for preparation of the canisters with spent fuel for shipping to the re-processing facility ay Mayak shall ensure:

- Non-exceeding the dose limits for personnel exposure;
- Safe temporary storage of canisters with spent fuel in the cooling pools;
- Observance of the paks npp safe operation limits;
- Minimal release of nuclear materials and radioactive isotopes from the canisters;
- Minimisation of radwaste generation.

The technology should be based on the following provisions:

- For assurance of safe conditions of transport, the canisters with spent fuel must be dried, filled with an inert gas and sealed;

- Prior to drying, the current lids suitable for water-filled canisters must be replaced with lids suitable for gas-filled canisters; these activities are to be performed in pit №1;
- During storage and transport of dried gas-filled canisters, catalysts for oxidation of hydrogen are not used;
- Drying of the canisters is performed without prior drainage;
- Drying, filling with an inert gas and sealing of the canisters is performed inside a shielded technological module installed into pit k1;
- Before loading into the tuk, the gas-filled canisters are stored in the cooling pool;
- All movements of the canisters with spent fuel between the pool and pit k1 are performed by the re-fuelling machine;
- For the preparation of canisters for transport, existing equipment used for removal of spent fuel from the cleaning tank should be used to the maximum extent possible (it may be upgraded to accommodate the new tasks).

The technology is technically feasible, with a sufficient necessary level of safety assured. There are currently no major insurmountable technical or legal obstacles standing in the way of transporting the failed fuel canisters for re-processing to Russia.

Simulation of handling operations with type 28 and 29 canisters at Mayak demonstrated that they can be processed using the plant's standard technology. The equipment and technology of the plant are ready to receive and re-process the failed spent fuel from Paks NPP.

REFERENCES

[1] TRAVERS, W. D., NRC Staff safety evaluation in support of reduced canister void volume 71-9200 (1987).

DEVELOPMENT OF AN ALTERNATIVE CORROSION INHIBITOR FOR THE STORAGE OF ADVANCED GAS-COOLED REACTOR FUEL

P.N. STANDRING[a], B.J. HANDS[b], S. MORGAN[b], A. BROOKS[b]
[a] Thorp Technical Department, Sellafield Ltd
[b] National Nuclear Laboratory
Sellafield
United Kingdom

Abstract

Sellafield Lt. currently stores AGR fuel in sodium hyrodxide dosed pool water to pH 11.5 to prevent susceptible AGR fuel from failing due to inter-granular attack. The exception to the above storage practice is Thorp Receipt and Storage (TR&S) where an AGR reprocessing buffer is stored in demineralised water as the expected storage durations were short term (up to 5 years). With the extended shut-down of Thorp, storage durations have increased and this has prompted a re-evaluation of the AGR storage regime in TR&S. The use of sodium hydroxide is not feasible due to a compatibility issue with aluminum components used in LWR storage furniture. The implementation process adopted by Sellafield Ltd in developing an alternative corrosion inhibitor for spent AGR fuel is outlined. The two stranded approach evaluates the impact of candidate corrosion inhibitors on fuel integrity and on plant and processes. The development studies in support of the fuel integrity strand are reported. Candidate inhibitors were first evaluated inactively in terms of their ability to arrest propagating corrosion, radiation stability, compatibility with aluminium and environmental impact. Sodium Nitrate was concluded to be the most promising inhibitor. Sodium nitrate was subsequently tested with active AGR brace material. These studies involved the use of bespoke test equipment and techniques. The studies demonstrated that propagating corrosion could be arrested using 10 ppm nitrate and showed that the resultant nitrate film required relatively high chloride concentrations to break it down over the study duration of 60 days. The development studies to date have provided the confidence that sodium nitrate has the potential to be an effective inhibitor for AGR fuel. The final phase of the fuel integrity strand involves a Lead Container Study using whole AGR pins. A staged approach is being adopted in the study programme where proceeding to a more onerous study is not progressed until positive results are concluded from the previous trial. This final stage of testing has recently been initiated and is expected to be completed during 2011.

1. INTRODUCTION

Sellafield Ltd (SL) is contracted to manage the lifetime irradiated Advanced Gas-cooled Reactor (AGR) fuel arisings from British Energy's seven reactors. The current planning basis [1] is a mixture of reprocessing (covering the planned life of the Thorp reprocessing plant) and a period of interim wet storage for the remainder of the fuel in the Thorp Receipt and Storage facility (TR&S). Interim storage could be up to 80 years and must be compatible with direct disposal acceptance criteria and projected repository availability.

AGR fuel elements comprise 36 stainless steel clad fuel pins, containing uranium dioxide fuel pellets that are held together by stainless steel braces enclosed in an open-ended graphite sleeve which acts as part of the neutron moderator. Normally 8 fuel elements (7 in the case of Dungeness NPP) are held together by a tie rod running through the central tube of each fuel element to make up what is referred to as an AGR stringer. After irradiation, the stringer is dismantled into individual fuel elements before being wet stored in fuel skips.

It has been well reported [2–6], that irradiated AGR fuel elements 1–5 of the original irradiated stringer are known to be susceptible to irradiation induced inter-granular stress corrosion cracking of the stainless steel fuel cladding and structural components. To inhibit this failure mechanism, AGR fuel is stored in pool water dosed with sodium hydroxide to pH 11.5. Sodium hydroxide was chosen as a result of a corrosion inhibitor development programme undertaken in the early 1980s and has been used since 1986 for the interim storage of AGR fuel at Sellafield. Operational experience to date indicates that fuel cladding perforation has been totally prevented.

The exception to the above storage practice is TR&S where an AGR reprocessing buffer is stored in demineralised water as the expected storage durations were short term (up to 5 years).

With the recent extended shutdown of Thorp, the AGR storage buffer in TR&S has had to be increased in order to sustain fuel receipts from reactor stations. This has prompted a review of the current storage regime for AGR fuel in TR&S. Whilst sodium hydroxide dosing of TR&S is the long term preferred option (as has been proposed for the interim wet storage of AGR in TR&S), dosing the whole of TR&S in the near term has been ruled out because of the potential for sodium hydroxide to react with aluminium components contained within Light Water Reactor (LWR) Multi Element Bottles (MEBs) which are co-stored in the facility.

The current paper out-lines the process in taking candidate corrosion inhibitors to full plant implementation. Only the development studies in support of spent fuel integrity will be discussed.

1. IMPLEMENTATION PROCESS

The process in taking a concept to full plant implementation is outlined in Figure 1. This comprises two strands: one strand evaluates the fuel integrity issues and other evaluates the potential impact on plant and processes.

2. FUEL INTEGRITY STUDIES

3.1. Fuel integrity - inactive studies

For contingency planning purposes, against, for example, the inability to remove redundant LWR MEBs in a timely manner, preliminary investigation of alternative corrosion inhibitors to sodium hydroxide were already in-hand; initiated in 2003. The investigation re-evaluated inhibitors (sodium borate, chromate, molybdate, nitrite, silicate, hydroxide/silicate and ammonium pertechnetate,) which had been studied as part of the original corrosion inhibitor development programme in the 1980s, and included the new materials sodium nitrate and high quality demineralised water. The candidate inhibitors were assessed in terms of radiation stability, compatibility with aluminium, and environmental impact. A short list of suitable inhibitors (demineralised water, sodium nitrate, sodium silicate and sodium silicate/sodium hydroxide) were subsequently subject to corrosion testing to demonstrate their ability to arrest propagating inter-granular attack. This was achieved through the application of an inter-granular attack (*iga)* corrosion sensor; a macroscopic simulation of a chromium depleted grain boundary.

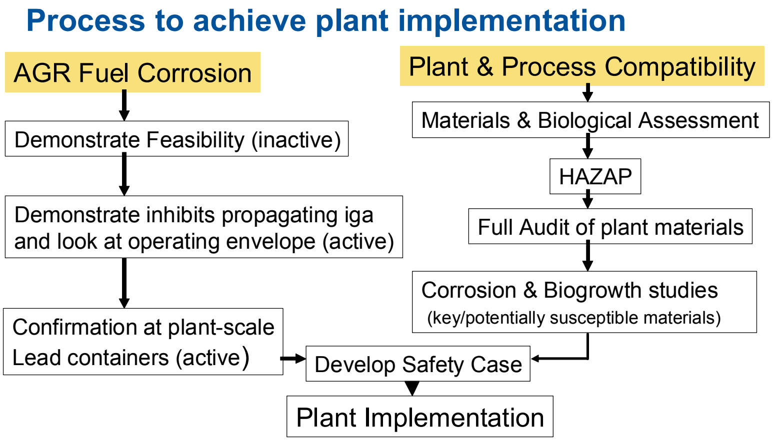

FIG. 1. The process from a concept to full plant implementation.

Based upon the test criteria, sodium nitrate at a concentration of 10 ppm was selected as the most suitable corrosion inhibitor for active testing. The basis of using 10 ppm nitrate stems from a previous study on the prevention of crevice corrosion of 304L stainless steel used in cooling circuits of reprocessing and waste treatment plants which concluded that a 5:1 ratio of sodium nitrate to chloride is required to inhibit corrosion. Worst case chloride ingress is taken as 2 ppm chloride which is derived from site salt disposition studies.

3.2. Fuel integrity — active studies (hot laboratory)

The candidate corrosion inhibitor (sodium nitrate) was subsequently taken forward for active studies in shielded cave facilities in the Windscale Laboratory at Sellafield. Corrosion studies were undertaken by monitoring the response of AGR brace material to changes in solution chemistry by the application of the zero resistance ammeter (ZRA) technique.

Brace material was chosen as previous inhibitor studies had used braces. Braces receive a higher dose than fuel pins for the same irradiation temperature; the principle that sensitisation increases with dose. Historical Post Storage Examination of braces had shown them to be more susceptible to corrosion than fuel pins. It was also found to be difficult to initiate corrosion on fuel cladding in electrochemical corrosion tests; probably due to the ability to select the right pin section. The use of braces avoids testing whole pins.

The scope of the active tests included:

- Demonstrating that propagating corrosion could be inhibited by sodium nitrate;
- Evaluation at two operating temperatures;
- Evaluation of the impact of increasing chloride concentration at constant inhibitor concentration;
- Evaluation of the impact of decreasing inhibitor concentration at constant chloride concentration.

The experimental set-up and innovative approach to fabricating the zero resistance ammeter (ZRA) electrodes and corrosion tanks is detailed in [7]. The set-up is briefly described, radial three strip sections of brace were cut using a diamond tipped coring tool and these were then spot welded onto a specimen holder. A brace sample and two sections of AGR cladding were then mounted into re-useable electrode holders. The fabricated electrode holders were then placed into tanks (each capable of holding 12 specimens) and the position of each electrode holder and sample identity was noted. Fig. 2 shows the set-up (individual pictures show the brace, ZRA electrode assembly and test tank arrangement).

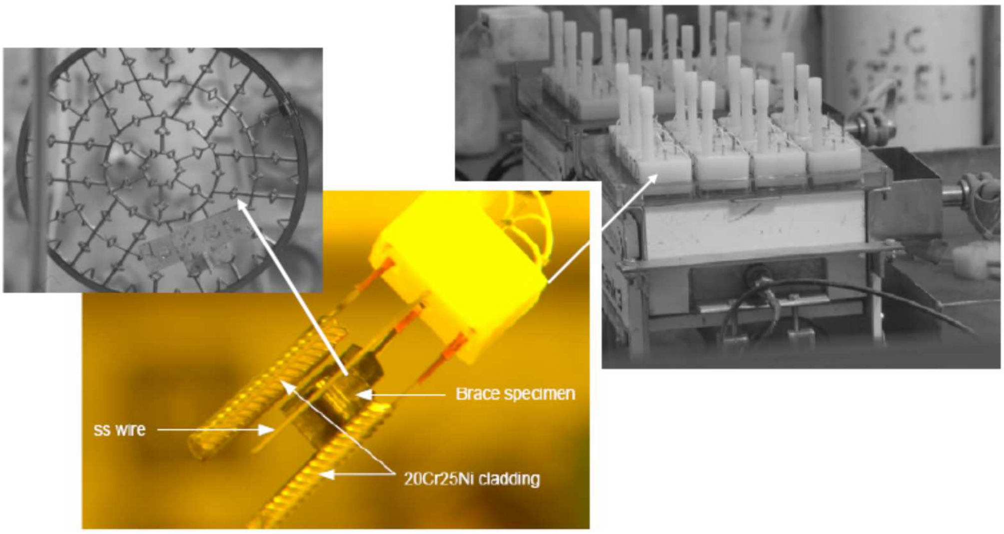

FIG. 2. The brace, ZRA electrode assembly and test tank arrangement.

Corrosion of the brace samples was monitored using two multi-channel ZRAs. An electrical connection between the ZRA and the brace specimens was established using a through-wall cable which connected to the tank lids which comprised a printed circuit board facilitating electrical contact to the brace and cladding electrodes via spring loaded pins in each electrode holder. An example of the multi-channel ZRA output for measured current is shown in Fig. 3.

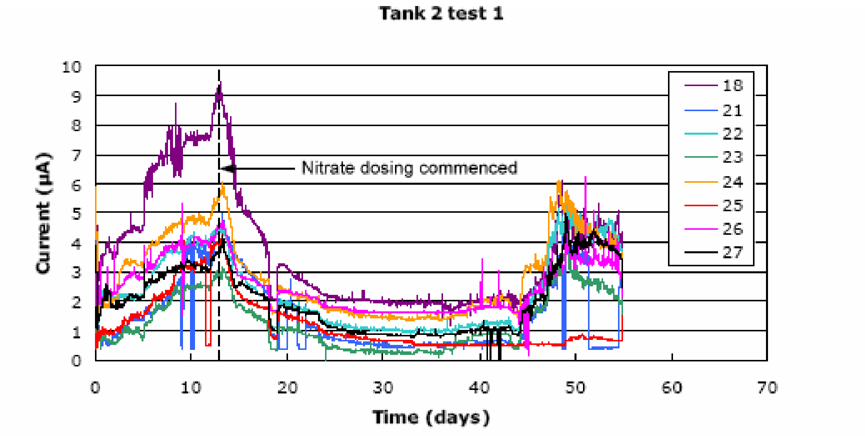

FIG. 3. Example of the ZRA output for measured current.

Test solutions were supplied by a constant solution feed from out-of-cave reservoirs. Peroxide was added to simulate the effects of solution radiolysis and carbonate was added to replicate the adsorption of atmospheric carbon dioxide. Tests were conducted over a period of 60 days.

For example, the test sequence adopted to demonstrate propagating corrosion can be inhibited:

- **Corrosion initiation** — 2 ppm chloride in demineralised water;
- **Corrosion inhibition** — Day 13 tank chemistry changed to 10 ppm nitrate + 2 ppm chloride in demineralised water;
- **Corrosion re-initiation** — Day 40 nitrate removed. 2 ppm chloride in demineralised water introduced. Chloride content progressively increased to 50 ppm in order to break down the protective nitrate film.

The ZRA output for this test sequence is shown in Fig. 4.

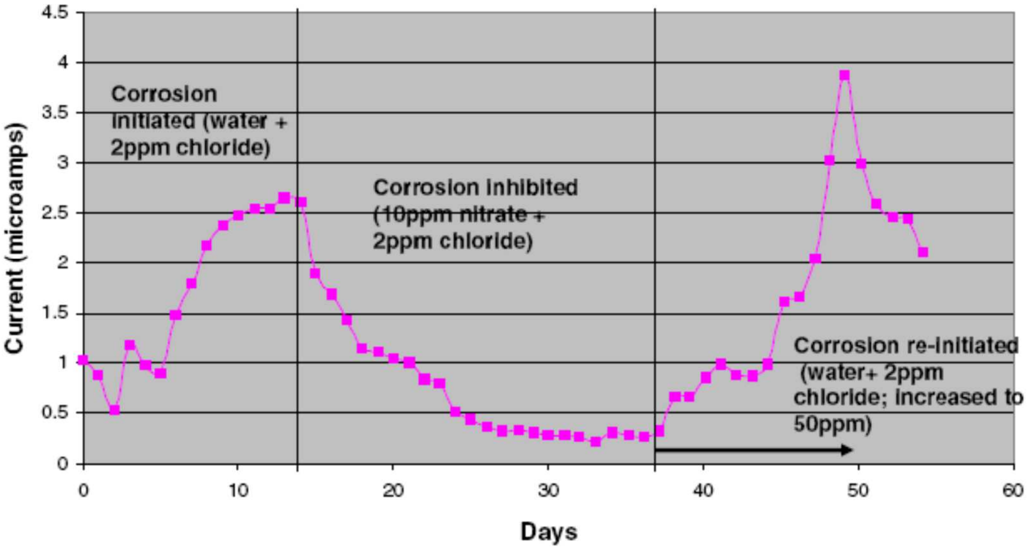

FIG. 4. The ZRA output for the test sequence adopted to demonstrate propagating corrosion.

Further testing has also demonstrated the effectiveness of the nitrate over the test period studied. There was no initiation of corrosion for either reducing the nitrate concentration (reduced to 1 ppm at a constant 2 ppm chloride) or increasing chloride at constant 10 ppm nitrate (chloride increased to 50 ppm).

To support the electrochemical studies, samples taken from the braces used in the tests have been subjected to materials analysis [Scanning Electron Microscopy (SEM), Strauss testing and grain boundary analysis] to confirm that the material being tested was sensitised. Strauss testing in a number of cases led to complete destruction of the sample. Fig. 5 shows the residues from one sample after testing and an SEM shows individual metal grains. These studies confirm that the brace material tested was severely sensitised.

The results from the active tests have provided the confidence that propagating attack will be inhibited by the application of sodium nitrate and passivation is likely to be achieved within 15 days of dosing. It also appears that the resultant nitrate film offers a high degree of protection which requires relatively high chloride concentrations in the short term to break it down.

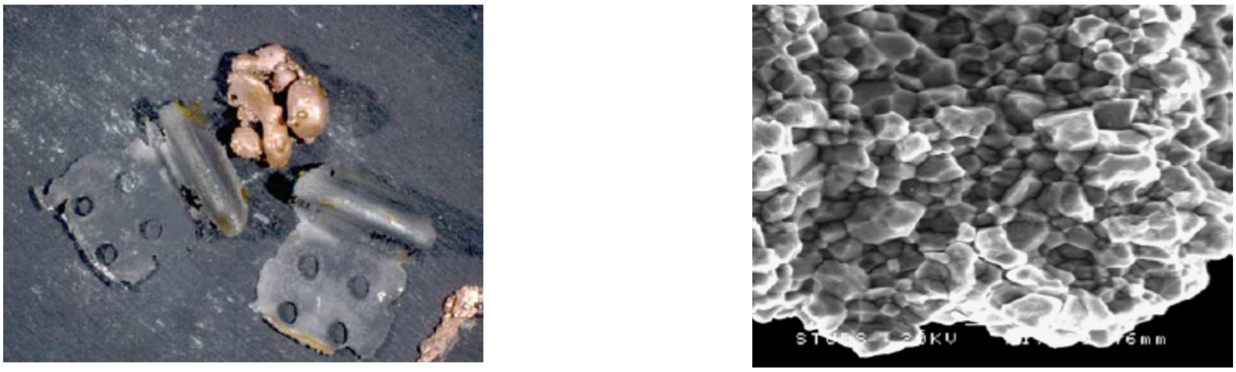

FIG. 5. Residues and SEM photo from one sample after Strauss testing.

3.3. Fuel integrity — active studies (lead container studies)

Fuel corrosion studies to date have provided the basis and confidence that sodium nitrate would act as an efficient corrosion inhibitor for AGR fuel and is unlikely to impact on its integrity if deployed. The next step in the implementation process is to confirm active (Hot Laboratory) study findings through a Lead Container Study; which utilizes whole AGR pins. In doing so, not only is the sample size increased significantly, but the following differences between braces and pins are addressed:

- Braces are made of 20:25:Ti Stainless steel c.f. 20:25:Nb Stainless Steel fuel pin cladding;
- Braces have crevices c.f. pins have no crevices;
- Braces levels of stress are minor c.f. pins.

A total of 5 lead containers will be used in the study. One container will be used as a control and the other 4 containers will be used (one for each study).

The main objectives of the four studies are to:

- Confirm that the introduction of sodium nitrate to 10 ppm has no impact on AGR fuel integrity;
- Confirm that the introduction of sodium nitrate to 10 ppm has no impact on known failed AGR fuel;
- Confirm that under nitrate dosing to 10 ppm a chloride excursion up to 2 ppm does not initiate fuel corrosion (applied to studies 1 and 2 after the initial objective is confirmed);
- Confirm that fuel that is undergoing propagating attack (fuel corrosion) is inhibited by the addition of sodium nitrate to 10 ppm;
- Confirm that the overall impact of establishing the recommended 10 ppm sodium nitrate passive regime has minimal impact on fuel that is undergoing propagating attack.

In studying these objections a staged approach will be applied whereby confirmation of the first objective is required before proceeding to the next objective.

The container set-up is shown in Figs 6.1. and 6.2. To facilitate routine sampling the containers are hard-wired or plumbed to a dedicated dosing and sampling rig at the pool side. Each container is fitted with four catalytic recombiners and a corrosion probe. To isolate the container water from the bulk pond water a nitrogen gas ullage is introduced into each container; this is maintained by a small nitrogen bleed.

FIG 6.1. Container wired to sampling rig.

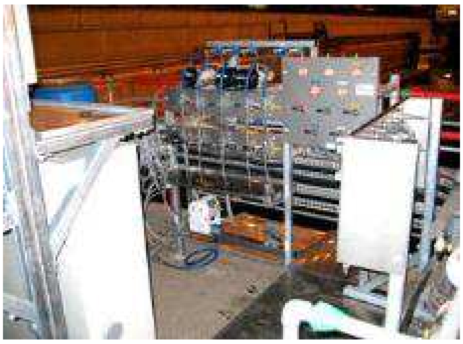

FIG. 6.2. Dosing/sampling rig.

The expected duration of the Lead Container Study is around one year in the first instance. As can be seen from the pictures provided above the study has recently been initiated and will report in 2011.

3. SUMMARY

Sellafield Ltd currently stores AGR fuel in sodium hyrodxide dosed pool water to pH 11.5 to prevent susceptible AGR fuel from failing due to inter-granular attack. The exception to the above storage practice is Thorp Receipt and Storage (TR&S) where an AGR reprocessing buffer is stored in demineralised water as the expected storage durations were short term (up to 5 years).

With the extended shutdown of Thorp, storage durations have increased and this has prompted a re-evaluation of the AGR storage regime in TR&S. The use of sodium hydroxide is not feasible due to a compatibility issue with the aluminum components used in LWR storage furniture.

The implementation process adopted by Sellafield Ltd in developing an alternative corrosion inhibitor for spent AGR fuel is outlined. The two stranded approach evaluates the impact of candidate corrosion inhibitors on fuel integrity and on plant and processes. The development studies in support of the fuel integrity strand are reported.

Candidiate inhibitors were first evaluated inactively in terms of their ability to arrest propagting corrosion, radiation stability, compatibility with aluminium and environmental impact. Sodium nitrate was concluded to be the most promising inhibitor.

Sodium nitrate was subsequently tested with active AGR brace material. These studies involved the use of bespoke test equipment and techniques. The studies demonstrated that propagating corrosion could be arrested using 10 ppm nitrate and showed that the resultant nitrate film required relatively high chloride concentrations to break it down over the study duration of 60 days.

The development studies to date have provided the confidence that sodium nitrate has the potential to be an effect inhibitor for AGR fuel. The final phase of the fuel integrity strand involves a Lead Container Study using whole AGR pins. A staged approach is being adopted in the study programme where proceeding to a more onerous study is not progressed until positive results are concluded from the previous study. This final stage of tesing has recently been initated and is expected to be completed during 2011.

ACKNOWLEDGEMENTS

In executing this work package the National Nuclear Laboratory was contracted to undertake the inactive inhibitor studies, active brace corrosion studies, build of the Lead Container sampling/dosing rig and plant compatability studies and Studsvik AB was contracted to evaluate the materials properties of AGR brace material.

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DAMAGED FUEL STORAGE AND RECOVERY - A CASE STUDY

D.L. FISCHER[*], N.C. IYER[*], R.L. SINDELAR[*], J.E. THOMAS[*], T.J. SPIEKER[**]
[*] Savannah River National Laboratory
Aiken SC
[**] Savannah River Nuclear Solutions
Aiken SC
USA

Abstract

Since the 1960s, spent nuclear fuel section test pieces and other severely damaged nuclear fuel sections from a variety of sources was placed into cans within large isolation canisters, termed Oversize (OS) canisters, for interim wet storage at the Savannah River Site (SRS) Receiving Basin for Offsite Fuel (RBOF). The water activity in the OS canisters had risen with time to extremely high levels (5.8E6 Bq/ml in canister A5) and the canisters also contained corrosion product sludge debris. The fuel cans from the RBOF OS canisters were destined to be removed and transferred to the SRS L-Basin to consolidate fuel storage at the site and complete the fuel deinventory of RBOF. A special underwater filter/deionizer apparatus was designed, constructed, and deployed to efficiently capture the cesium from the OS canister water prior to opening the OS canister and to prevent it from dispersal into the general RBOF basin water and mitigate high radiation levels above the water. New OS canisters for L-basin were designed to hold the cans of the damaged fuel from RBOF. The new OS canisters included updated features for ease of handling and storage in racks in L-basin. The new canisters included a modified vent path from the canister to the general basin water to allow hydrogen to safely escape from the corroding fuel while containing high-activity water with the canister. This case study will relate experiences with containment of damaged fuels in pool storage and mitigation of effects of any potential cesium release into the water.

1. INTRODUCTION

Spent nuclear fuel (SNF) from research reactors, sections of material production reactor fuel, and test pieces and failed fuel from early commercial reactors, had been stored at the U.S. Department of Energy's (DOE) Savannah River Site (SRS) since the early 1960s decades. The fuel was originally stored in the Receiving Basin for Offsite Fuel (RBOF), a 600,000 gallon stainless steel-lined water pool designed for spent fuel storage. These fuel sections included various fuel compounds, cladding materials, and geometries.

Some damage was the result of experimental activities and destructive examinations; other damage may have occurred in the reactor or from problems encountered in packaging, handling, and transportation, or degradation during storage. SNF damage includes severely failed cladding, exposed fuel meat, sectioned test specimens, over-heated elements, dismantled assemblies, and assemblies with lifting fixtures removed. The cut/damaged fuel was placed into cans and these cans were loaded into isolation canisters, termed Oversize (OS) canisters. The use of Oversize canisters to isolate highly damaged fuel that would contribute significantly to basin radiological activity is a good system to enable its underwater storage. This system has been used successfully for many decades at SRS to safely handle and store damaged fuel.

The SRS transitioned to using the L-basin for receipt and storage of aluminum-based, aluminum-clad research reactor fuel beginning in the mid-1990s. The L-basin is a 3,375,000 gallon vinyl-coated concrete basin that was formerly a temporary cooling pool for the site's production reactors. Fuel with intact cladding, or breeched research reactor aluminum-based fuel with aluminum cladding and has only slow minor release of cesium through corrosion and is readily removed by the general deionization system of the storage basins [1], is stored in racks in the L-basin.

The site decision to consolidate its total inventory of spent nuclear fuel into the L-basin in the late 1990s required recovery of the damaged fuel from the OS canisters in RBOF. A challenge to the damaged fuel recovery and repacking activity for storage in L-basin was the desire to avoid release of the highly contaminated water from the OS canisters into the RBOF water. This paper reviews the options for the pool storage of damaged fuel in oversized canisters and summarizes the design, development, and deployment of a special underwater filter/deionizer system to remove the activity from the water in the OS canisters. The fuel cans were placed into new design OS canisters for storage in L-basin.

2. DAMAGED FUEL STORAGE IN WATER BASINS

Isolation canisters for damaged SNF are designed to allow underwater storage of the highly damaged fuel while containing the radionuclide activity that can be released from it. Containing the released activity is important from the standpoint of radiological protection of basin operations personnel and reducing waste generated by more frequent regeneration/replenishment of basin deioinizer resin beds. A common design feature in the SRS-design OS canisters was a J-tube in the lid of the canister (Figure 1). J-tubes are similar to a P-trap on a household sink. Gases released from damaged SNF build up at the top of the can, separating the internal water environment from the main basin. The open design of the original J-tubes made them susceptible to silt buildup and pluggage. Valved couplings were added to avoid this problem, but this restricted the free flow of gas from the canisters. To minimize the potential for pluggage of the tube with debris, changes in the J-tube design were made (Fig. 2).

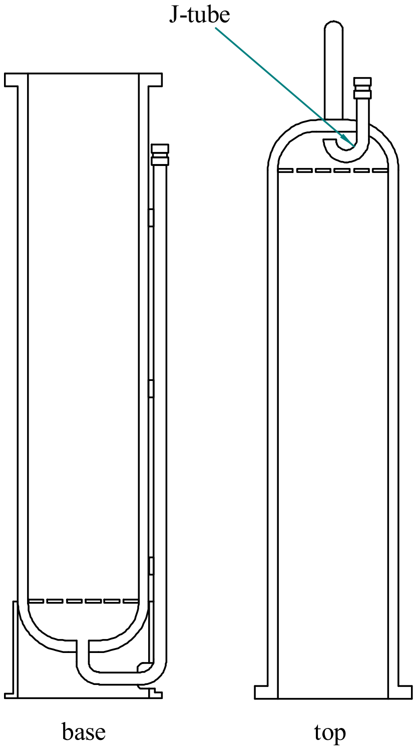

FIG. 1. One form of J-tube in a damaged-fuel storage canister.

Figure 2 shows that, with the opening at the top of the canister, very little gas build-up is needed to separate the two water environments. The canister designed to transfer fuel from RBOF incorporated a modified version of the J-tube.

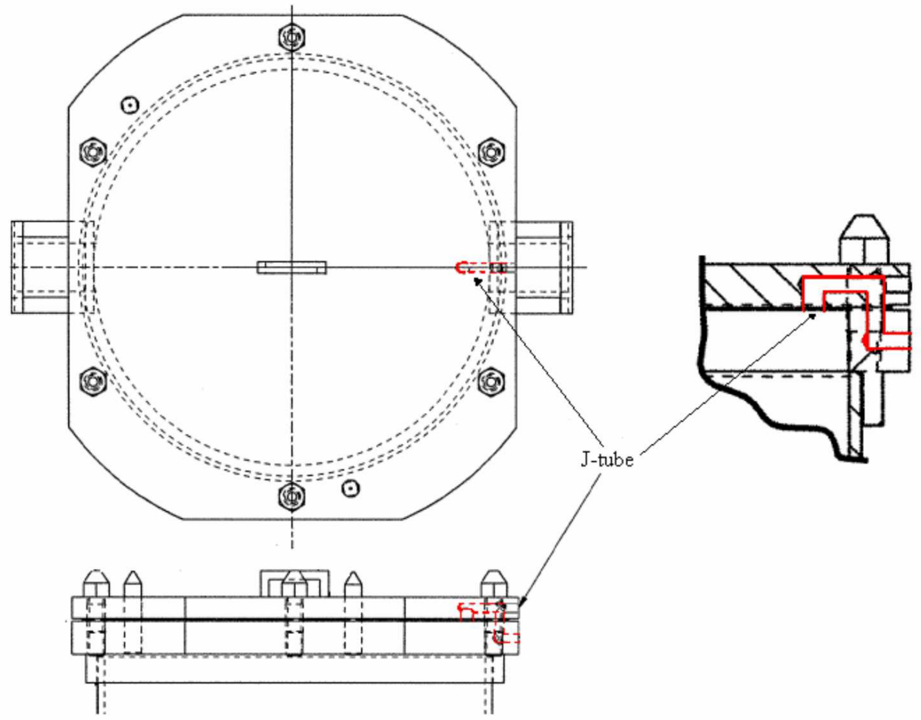

FIG. 2. Later J-tube design on damaged-fuel storage can.

SRS chose isolated storage in water for the significantly damaged and cut fuel, separating it from the remainder of the basin water by loading it into large water-filled cans. The method of storing damaged fuel underwater was to place fuel pieces in small diameter cans fabricated from either aluminium or stainless steel which were grouped with others and placed in larger aluminium vessels referred to as over-sized (OS) cans (Fig. 3). OS cans were 3.96 m tall right circular cylinders with an interior diameter of 0.33 m, made in two halves with flanges in the centre of the long axis. They were designed with an exterior access tube from the bottom of the vessel and a j-tube at the top. Gas emanating from the fuel could be vented out of the OS can through the j-tube if desired. Couplings were affixed to the end of both of these tubes, isolating the contents of the OS cans from the basin, but providing a path to flush water through the vessel or take water samples.

2.1. Basin closure activities

The above mentioned containment system functioned effectively during the decades of the 70s, 80s 90s and into the 2000s. However, the storage basin where these cans were stored was deemed excess and scheduled for closure after transferring its contents to another SRS storage basin. The plan for transfer included repackaging the severely damaged fuel into containers that would fit in existing fuel racks at the destination facility.

Repackaging damaged fuel required the OS cans to be opened, which could quickly mingle basin water with the water in the cans. Water samples were taken from inside some of the OS cans to provide information on radioactivity. Radiation emitted from the 60 ml samples was significant (10^6 disintegrations per minute) indicating the water inside some of the OS cans would require treatment before releasing it or mingling it with the general basin water. Three primary options seemed readily available, all of them made use of the inlet and outlet tubes on the OS cans. Contaminated water from the OS cans would have to be stripped of the bulk of its radioactive isotopes through ion exchange resin before releasing it to the basin. The RBOF basin was equipped with a large deionizer system which offered options for treatment of the OS can water. Altogether, three options were considered.

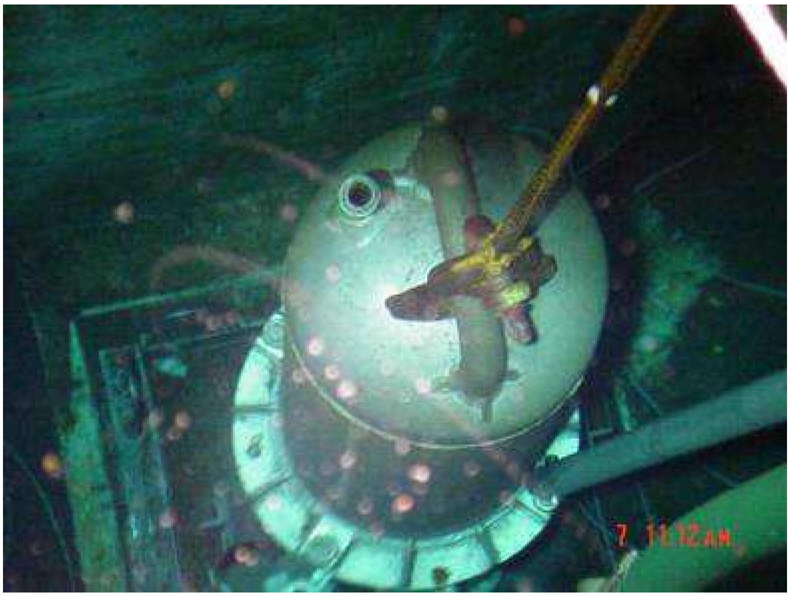

FIG. 3. An oversized aluminium storage can used for damaged-fuel (at -9.14 m) in RBOF.

These were:

(a) Whole-basin decontamination using the RBOF ion-exchange system;
(b) OS can flushing, using the RBOF ion-exchange system in a partial bypass arrangement;
(c) OS can flushing with a small dedicated ion exchange system.

Repackaging would impact the closure schedule if any of the options took very long to implement, so time was an important element in the considerations. The first option required processing at least the entire basin volume, resulting in large quantities of liquid waste. Liquid waste generated by the RBOF deionizer (from ion-exchange regeneration and flushing) was cited by basin operations personnel as a limiting factor since their liquid waste allotment for the year was nearly consumed. However, this option did not require any equipment additions and would not delay the work. The main objection to this option was that it would contaminate the basin with radioactive cesium, causing the activity limit to be exceeded and therefore a potential delay or curtailed operations.

The second option is similar to the first except that the RBOF ion-exchange system would be equipped with an open suction in the general basin as well as drawing basin water through the OS can. This would allow the flow rate through the OS can vent-tube to be reduced to a reasonable rate but would require some modification to the facility equipment and presented the possibility of producing significant liquid waste.

The third option was to acquire a small submersible, low-flow ion-exchange system to treat only the water in the OS cans (plus a small amount of flush water). By using enough resin to capture the cesium in this water, then disposing of the resin as waste, no liquid waste is produced and only a small resin column is produced as waste. This option was chosen and a list of functional criteria was generated.

2.2. Underwater deionizer design

The historical archival data related to basin conditions and resins successfully used to control water chemistry was assimilated. A conservative estimate of the quantity of dissolved cesium was made resulting in determination of the total resin requirements. The RBOF fuel storage basin water was clear and clean and its activity was limited to 2 mRem/hr at the surface. The deionizer capacity and efficiency had to be adequate to capture 1000 Ci (11.5g) of ^{137}Cs from the oversized cans, processing a minimum of 9 m^3 of water. The amount and size of particulate which might be present in the oversized cans was unknown but particles were presumed to be present from cut-fuel deterioration inside the cans. A water sample from one oversized can was drawn, indicating a conductivity of 200 μS/cm and an activity of 5.8×10^6 Bq/ml. Based on this information and the records of oversized can contents, a list of functional design requirements was established. These included:

(a) System must remove ≥90% of ^{137}Cs (1000 Ci ≈ 11.5 g);
(b) System components must be remotely serviceable, remotely replaceable, or redundant;
(c) A pre-filter must be present to remove particulate prior to ion-exchange;
(d) Provide simple flow measurement;
(e) Provide fluid sample ports upstream and downstream of the ion-exchange column;
(f) Design pressure boundaries (where applicable) to ASME BPV and B31.3;
(g) Design system to operate under 9.1 m of water;
(h) Ion-exchange column and filters must fit in a designated container;
(i) Supply tools to operate/maintain the system remotely;
(j) Replacement parts must be available within 2 days or a backup system must exist.

2.2.1. Selection of resin for the ion exchange column

An ion exchange resin is an insoluble media (e.g. small beads of polymeric material) with functional or active group ions that are loosely attached to the backbone polymeric structure of the resin. The functional group is readily replaced with ions present in the water. That is, the impurity ions removed from water are trapped with the release of ions from the ion exchange media. Ion exchange resin systems are ideal for removing dilute impurities from the water in the primary and spent fuel basin systems.

The resin is in the form of small beads, typically 1–2 mm in diameter. A small bead diameter provides a high surface area and overall capacity in a given volume. However, the small diameter beads provide a greater resistance to flow in a bed or column system. Beads are retained in their containers with screens above and below the resin beds in the container. Normal fluid flow is top down to keep from fluidizing the resin beads and thereby decreasing their effectiveness.

Several parameters of a resin must be considered in the design of an ion exchange resin column system for a reactor water system. Important design parameters are:

- Species and amount of species to be removed;

- Flow rate per volume of resin;
- Minimum height of resin bed;
- Equivalency (capacity) of the resin media;
- Allowable temperature range of operation of the resin media;
- Allowable pH range of operation of the resin.

The above parameters were considered in the selection of the resin media and design of the resin column for the underwater filter/deionizer. A single gel type strong acid cation resin in the hydrogen form, Resintech CG8-H was selected for the removal of cesium from the canister water. This resin is a typical co-polymer matrix is the polystyrene polymer with the divinyl benzene polymer added to crosslink the polystyrene to provide structural stability. SRS experience has shown this resin to provide excellent removal and retention of cesium-137 the basin water for the pH range of the basin water. This resin is highly selective for cesium if the hardness species Ca and Mg have been removed.

Ion exchange resins have an equivalency that defines their capacity to incorporate ions from the water. An equivalency calculation was performed to calculate the amount of resin needed to capture the cesium ions from the water in the OS cans onto the resin. A total of 28 liters of CG8-H was estimated to provide sufficient capacity to complete the OS flushing campaign.

2.2.2. Design of deionizer column and system

Once the resin volume was identified, column proportions were set, followed by the range of flow rates to keep the operation within the efficiency range recommended by the resin manufacturer. The desired flow rate guided the remainder of the design. As a check on the column/resin combination, a full-scale prototype column was built and loaded with resin. A solution of non-radioactive cesium carbonate (Cs_2CO_3) was prepared by dissolution in deionized water. The prepared stock solution contained 35.4 g of Cs at a concentration 56.7 mg/l. Pumping this solution through 28.3 l of resin in the recommended flow range (3 sets of tests) resulted in no measurable Cs in any of the 15 effluent samples. Total cesium supplied to, and captured in, the resin was equivalent to 3080 Ci of ^{137}Cs.

The required parts of the submersible deionizer in its simplest form were: a pump and motor, filter, and an ion-exchange column (Fig. 4). These had to be replaceable from the surface or have a spare mounted and ready for operation. The space allocated for the system was approximately 1 m^2 on a basin shelf at 4.6 m below the water's surface. The cation resin volume for the system bracketed system flow between 7.6 and 37.9 lpm to achieve the greatest ion exchange efficiency. The remainder of the system was sized accordingly. Positive displacement gear pumps (redundant) were powered by non-lubricated air motors, allowing variable speed operation and protection against locked-rotor situations. The water-lubricated pump(s) drew water from the oversized can(s) through a 100 micron wire mesh filter. The filter cartridge mesh size was chosen to protect the pumps and keep material from plugging the resin column. The design was checked for code compliance, fabricated, and fully tested before delivery to RBOF.

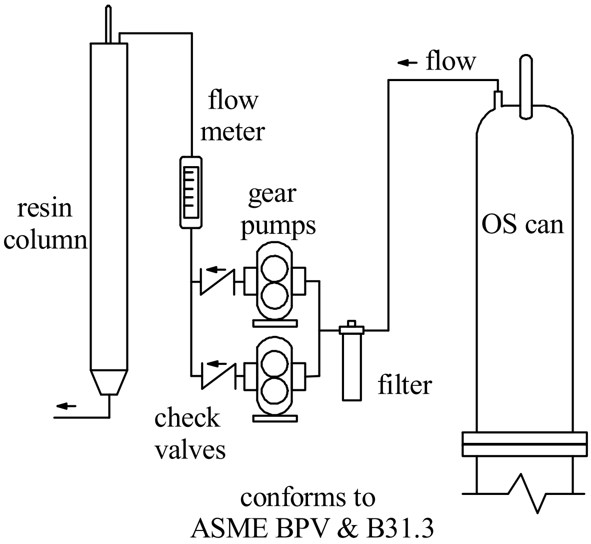

FIG. 4. RBOF underwater deionizer sketch.

The redundant pumps and their corresponding motors were fixed to a 0.71 × 0.91 m skid and plumbed together with stainless steel pipe, having a common suction from the filter. Each pump was followed by a check valve before be integrated with a common flow meter. The check valves allowed either pump to operate independently, without recirculating water to the suction, or simultaneously, if desired. The filter housing sat in a notched tubular receptacle which kept it from rotating and allowed it to be removed and replaced easily as a unit. The ion exchange column was aligned with its discharge pipe and held in position with parallel vertical guide rails. Other than the fixed piping on the skid, all liquid transfer was accomplished with reinforced rubber hose.

Plumbing that required remote make-and-break connections was equipped with double-shutoff automatic (push-to-connect) quick-disconnect fittings, allowing remote removal and replacement of the filter housing and the resin column, as well as inlet and outlet hoses. These required only 18 Newtons of downward force to connect and a light upward pull on the collar (with a special forked tool) to release. The important exception was the lower end of the ion-exchange column which would be very difficult to see or reach with tools. The latching ring was machined off of the male quick disconnect (nipple) mounted on the skid, allowing the ion-exchange column to be lifted without having to unlatch the mechanism. The weight of the column held the connection in place during operation. Figure 5 shows the underwater deionizer as it was positioned in the SRS RBOF basin, 4.6 m below the surface of the water.

3. LOCALIZED WATER ACTIVITY REMOVAL

The SRS Spent Fuel Project organization personnel executed OS canister flushing using the submersible deionizer with excellent results. Oversized cans consisted of 2 sections with a pipe connection to the bottom half and a j-tube vent at the top of the upper half. The j-tube vent was connected to the deionizer allowing low activity basin water to be drawn into the storage can with pump suction to bring high activity water into the deionizer (Fig. 5). Some of the storage can's inlets were restricted or plugged with particulate in which case the OS can flange was loosened to allow basin water to be drawn into the can through the flange opening for purging.

FIG. 5. RBOF underwater deionizer with empty oversized can halves.

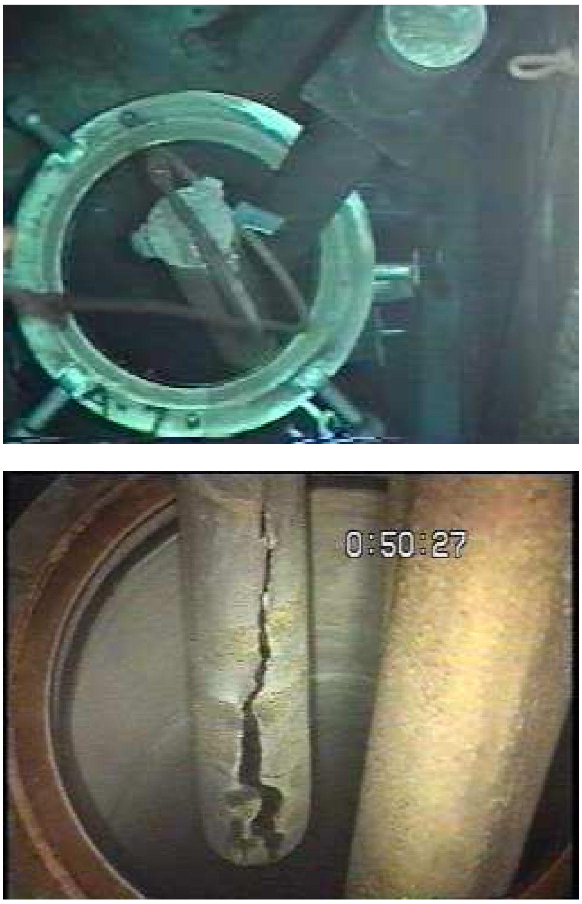

FIG. 6 RBOF OS can lower half/contents after flushing and after removal of the lower half contents.

Oversized can water activity was monitored using a submerged probe attached to a radiation monitor. Note that dose rates in the general basin water never exceeded the limit of 0.02 mGy/hr. The internal volume of each OS can was approximately 416 l (litres) and flow through the ion-exchange resin was maintained at 15.1 lpm. Table 1 below is a summary of water activity measurements at the deionizer column inlet and outlet. OS cans are listed in the order they were processed. Numbered cans missing from the table did not require flushing. Fig. 6 show some oversized can contents. The deionizer removed cesium from OS can water allowing damaged fuel to be moved into more modern stainless steel storage containers with an improved j-trap and designed to fit the racks in the destination (receipt) facility.

TABLE 1. RADIOACTIVITY BEFORE AND AFTER ION EXCHANGE AT DEIONIZER INLET/OUTLET

OS can	Initial Curie content (estimate Cs-137 in water)	RO7 initial reading	RO7 final reading (1 Gy = 100 rad)	Flush duration	Contents
A3	1.64	0.14 mGy/hr	0.01 mGy/hr	1 hour	1 FEC
A1	81	0.29 mGy/hr	0.01 mGy/hr	1 hour	1 FEC
A2	98	4.1 mGy/hr	0.02 mGy/hr	1.5 hours	1 FEC
A6	151	3.2 mGy/hr	0.04 mGy/hr	1.5 hours	2 FECs
A7	121	2.9 mGy/hr	0.02 mGy/hr	2 hours	2 Z cans 1 4” can
A5	297	24 mGy/hr	0.1 mGy/hr	3 hours	4 Z cans

FEC = fuel element container

4. SUMMARY

The submersible deionizer, designed and built at the Savannah River National Laboratory (SRNL), captured dissolved radioactive cesium from storage containers while breached fuel assemblies were repackaged at the Savannah River Site (SRS). The system generated no liquid waste and maintained the fuel storage basin radioactivity levels within limits. The deionizer resin would not be regenerated, but retained in its container and stored in a fuel storage container identical to those used to transfer and store the damaged fuel. Underwater measurements of radioactivity made at the red line near the resin column inlet indicated >60 mGy. This indicated that estimates of dissolved cesium were conservative and that the resin efficiently removed it from OS can water.

ACKNOWLEDGMENTS

The work and cooperation of many SRNL personnel and SRS Spent Fuel Projects colleagues is appreciated. The present work was supported through Contract DE-AC09-08SR 22470 with the U.S. Department of Energy.

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PROPOSALS ON AMB SNF MANAGEMENT

V. SMIRNOV
Research and Development Company "Sosny"
Moscow
Russian Federation

Abstract

The first commercial AMB reactors at the Beloyarsk NPP were shutdown in 1981 (the AMB-100 reactor) and in 1989 (the AMB-200 reactor). The SNF was discharged from the reactors and is presently stored in the cooling pools at the Beloyarsk NPP and in the storage pool at PA “Mayak”. The peculiarities of the AMB SFAs are as follows: a variety of ~ 40 types of fuel compositions, big dimensions, a complicated SFA configuration. The main problems for the AMB SNF handling at the Beloyarsk NPP are an unsatisfactory state of the fuel rods, corrosion of the SFA canisters and the cooling pool lining made of carbon steel sheets. At PA “Mayak”, all AMB SFAs are stored in stainless steel canisters. A special facility for pre-reprocessing preparation of the AMB SNF is planned to be constructed at the PA “Mayak”. The problem of the AMB SNF management is being resolved in two parallel directions: (1) ensure safe storage of the AMB SNF at the Beloyarsk NPP as a top-priority urgent task, (2) develop a technology and equipment for the AMB SNF reprocessing at the RT-1 plant of PA “Mayak”. Results of the work are addressed in the paper.

1. INTRODUCTION

The first AMB power reactors at the Beloyarsk NPP (BNPP) were shutdown in 1981 (the AMB-100 reactor) and in 1989 (the AMB-200 reactor). The SNF was discharged from the reactors and is presently stored in the cooling pools at the Beloyarsk NPP and in the storage pool at PA “Mayak”. The specific features of the AMB SFAs are as follows:

- A variety of ~40 types of fuel compositions;
- Large overall dimensions (the length of a sfa is ~13 m);
- A complicated sfa configuration.

At the Beloyarsk NPP the SNF is stored in cartridges made of carbon and alloyed stainless steel. The main problems in the AMB handling are heavy damage to the fuel rods, corrosion of the SFA canisters and the cooling pool lining made of carbon steel. At PA “Mayak”, all the AMB SFAs are stored in stainless steel canisters.

The main options for the AMB SNF management are:

(1) Shipment for temporary storage with subsequent solution of the reprocessing problem;
(2) Shipment for long-term storage with subsequent disposal;
(3) Cutting and canistering at the NPP, shipment for reprocessing to the PA” Mayak”;
(4) Delivery of SFAs to PA” Mayak”, cutting and reprocessing.

The disadvantages of dry storage of the AMB SNF are:

- Lack of justification for the safe transport and storage of assemblies because of the nuclear and radiation safety problems and hydrogen accumulation during storage and transport;

- Large dimensions and high cost of the ventilated storage;
- Lack of experience and regulating documents;
- Complication of the future reprocessing due to the ongoing SNF degradation during its storage.

As a result, a decision was made to reprocess the AMB SNF at Mayak.
It is supposed that a special building will be built at Mayak for preparation of the AMB SNF for reprocessing.

The problem of the AMB SNF management is being solved in two parallel ways:

- Ensuring safe storage of the AMB SNF at the Beloyarsk NPP as a top-priority and urgent task;
- Development of the technology and equipment for the AMB SNF reprocessing at the RT-1 plant of PA "Mayak".

For the purposes of realization of the AMB SNF reprocessing, the following problems are being solved:

(1) Development of the chemical reprocessing technology for all types of the AMB SNF;
(2) Design of the building at PA "Mayak" (cutting and containerizing facility — DCF) for the pre-reprocessing preparation of the AMB SNF, including the SNF stored at Mayak and the fuel stored in cartridges in case of their delivery from the Beloyarsk NPP;
(3) Development of two technologies for the SNF transport from the Beloyarsk NPP;
(4) According to one of them, SFA cartridges will be transported from the Beloyarsk NPP to Mayak and then cut at the DCF;
(5) According to the second technology, the cartridges and SFAs will be cut directly at the Beloyarsk NPP using the Complex of Equipment for Safe Cutting (CESC).

2. CHEMICAL REPROCESSING OF AMB SNF

Investigations carried out several tens years ago by the VNIINM experts demonstrated that, in principle, the AMB SNF could be reprocessed by the classical PUREX process, but that work was not tied to the RT-1 plant technology.

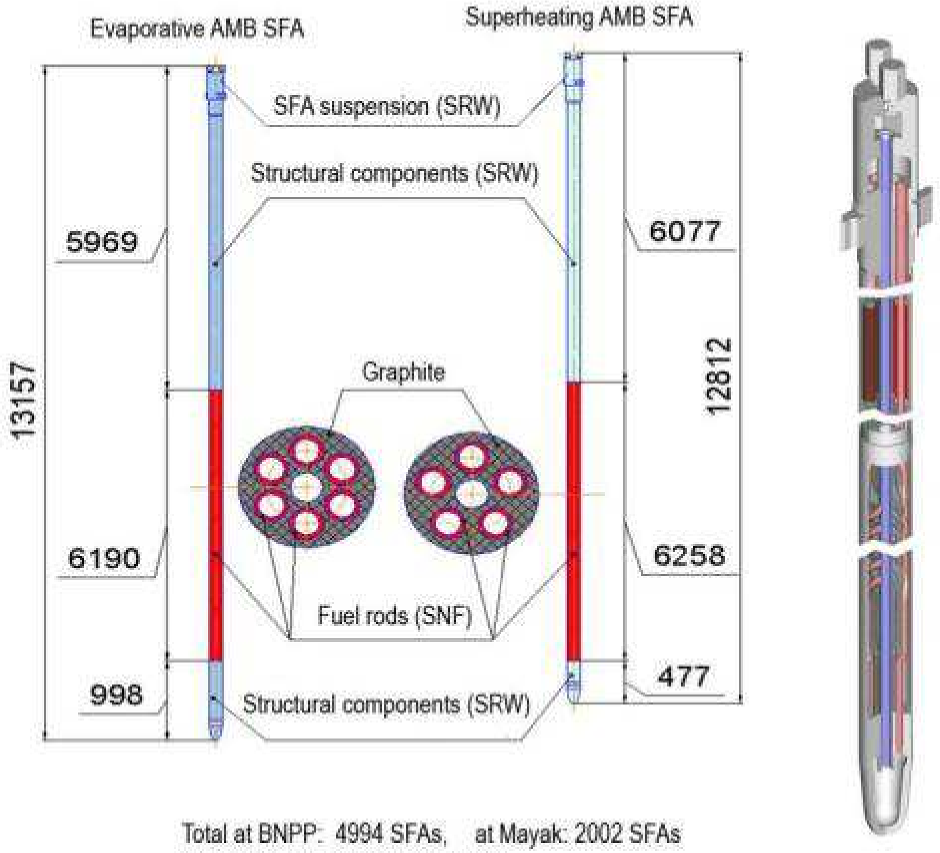
Therefore, reprocessing of spent AMB fuel at RT-1 is feasible, but there are a number of problems that need to be addressed at the stage of the scientific research. One of such problems is its transportation and cutting into fragments. The compositions of the AMB SNF vary to a great extent. The fuel composition may contain from 1% to 20% of ^{235}U and may consist of uranium oxide, metallic uranium, magnesium, calcium and molybdenum. For each fuel type, safe conditions of dissolution must be carefully chosen to prevent formation of explosive hydrogen concentrations.

A significant portion of spent AMB fuel also contains from 3– 9% of metallic molybdenum, dissolution of which may cause precipitation of both molybdenum and zirconium molybdate, which can entrap plutonium and cesium.

Problems may also occur during the evaporation and vitrification operations, which are part of the liquid radwaste processing. The spent AMB fuel reprocessing technology must be well built into the existing RT-1 processes. At present, the development of the chemical part of the process is nearing completion and there is an absolute certainty in its implementation at Mayak.

3. AMB FA CHARACTERISTICS

Design of the AMB SFA is shown in Figure 1.



Total at BNPP: 4994 SFAs, at Mayak: 2002 SFAs
Initial enrichment: from 1.5 to 21%
Average burnup: 15 MWd/kgU

FIG. 1. Design of the AMB fuel assembly.

The SFAs are stored in tubes in 17- and 35-seat cartridges (bundles) K-35 and K-17, respectively. All 95 cartridges of K-35 and 6 cartridges of K-17 are made of stainless steel; 139 cartridges of K-17u are made of carbon steel. At present all carbon steel cartridges are loaded into the thin-walled stainless steel canisters.

4. PREPARATION OF SPENT FUEL FOR SHIPMENT TO MAYAK

The main principles of the proposed technology for spent AMB fuel management.

By the first option all the SNF canisters are delivered to Mayak for reprocessing. There, at the DCF, the assemblies are cut into fragments and the fuel part is loaded into the thin-walled leak-tight canisters in which it is sent for reprocessing.

By the second option the following operations are performed:

(1) Separation of the fuel component from solid radioactive waste at the Special Cutting Facility (SCF) in the appropriate cooling pool;
(2) Loading of spent fuel into canisters meeting the Mayak requirements;
(3) Use of the TK-6 casks for transport of canisters from Beloyarsk NPP to Mayak;
(4) Reprocessing of canisters containing the SNF at Mayak after loading the canisters directly through the cooling pool (without the use of the DCF);
(5) Loading of solid radwaste into containers and solving the problem of its handling.

Maybe a decision will be made to use both options of the AMB SNF transport to Mayak.

The design concept:

- Meet the requirements of the NPP and nuclear fuel cycle (SNF) standards simultaneously;
- Adaptation of new production into the existing building at the Beloyarsk NPP of the first stage of construction;
- Own infrastructure to prevent the additional loading on the existing NPP systems for the basic production process.

In accordance with the concept all the work is subdivided into the following tasks:

(1) Development and implementation at Mayak of the chemical technology for the AMB SNF reprocessing;
(2) Investigations in support of the design solutions and safety analysis of the work;
(3) Design of tools, equipment and supporting systems;
(4) Manufacturing of the equipment and tools mockup;
(5) Final study of the technical decisions on the equipment and tools using the mockups. Personnel training;
(6) Preparation of the necessary technical, organizational and licensing documents;
(7) Fabrication of the basic equipment and tools;
(8) Mounting and commissioning work at the Beloyarsk NPP in accordance with the layout (Fig. 2);
(9) Practical work;
(10) Decommissioning of the equipment involved in the practical work.

5. RESULTS OF INVESTIGATIONS IN SUPPORT OF THE FIRE AND EXPLOSION SAFETY OF THE LEAK-TIGHT CANISTERS CONTAINING CUT AMB SNF FROM THE BELOYARSK NPP

Based on the analysis of the experimental data, the volume fraction of hydrogen accumulated for 5 years in the canister with cut and dried AMB fuel at the Beloyarsk NPP (OM9 in magnesium matrix) is estimated at less than 0.15% that confirms the fire and explosion safety of the package. The hydrogen accumulation rate for fragmented fuel under the water layer is 2–3 orders of magnitude higher than that for the dried SNF. Therefore, the explosive hydrogen concentration (4%) in the canister with the wet leaky SNF is formed in a day after its storage!

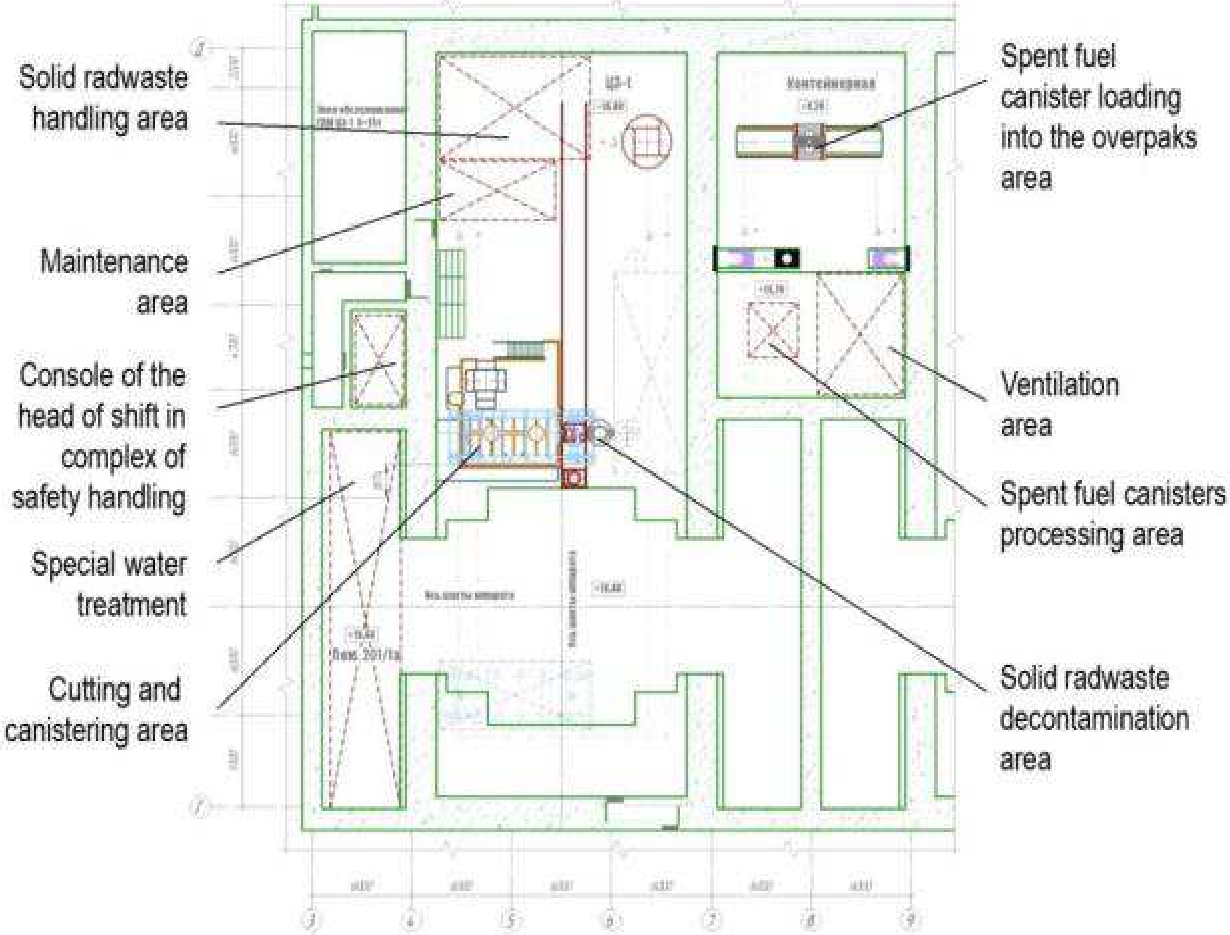


FIG. 2. Layout of the equipment at the Beloyarsk NPP.

It is assumed that the SFAs at the Beloyarsk NPP will be cut in water using the equipment located in the NPP hall (Fig. 3).

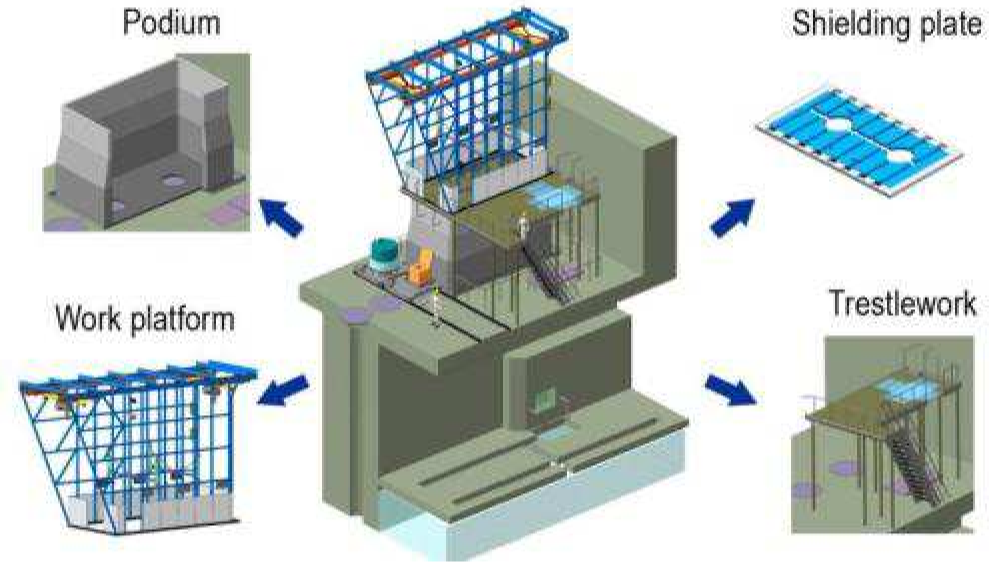


FIG. 3. View of the facility for cutting the cartridges and loading the SNF into canisters.

Fig. 4 schematically shows movements of cartridges containing the SFAs, their fragments as well as solid and liquid radioactive waste.

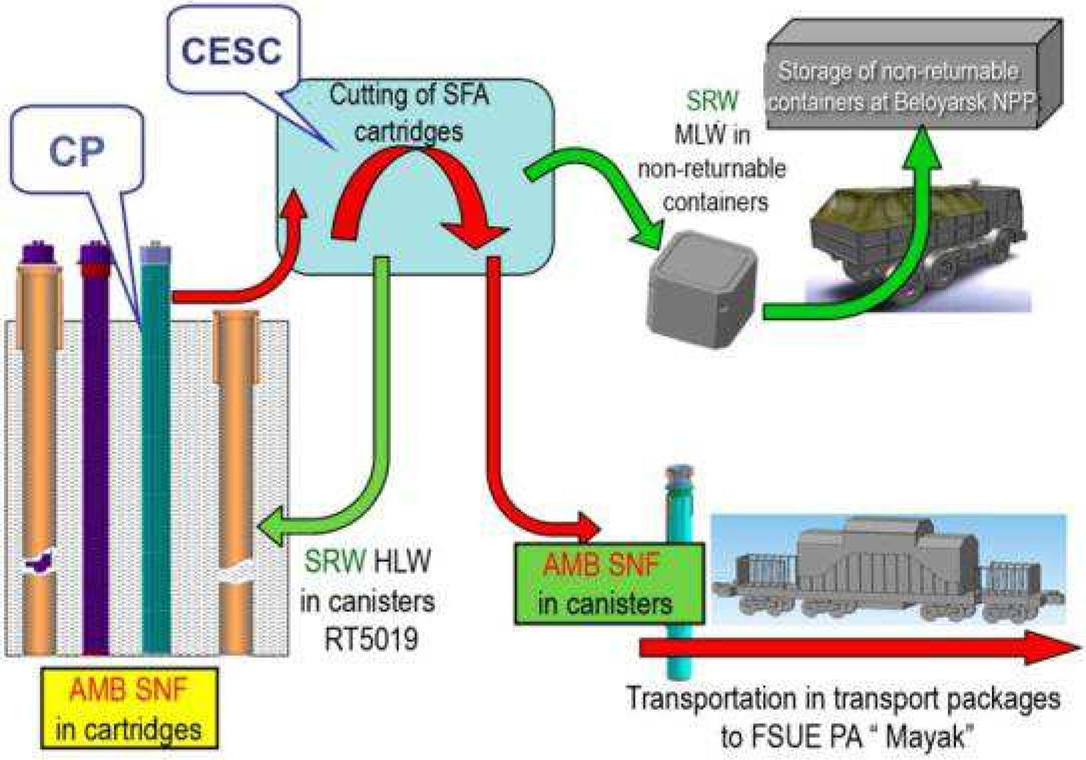


FIG. 4. Handling of the AMB SNF at the Beloyarsk NPP.

The DCF design principles (Figs 5 and 6):

1. Cutting the SNF using the water technology similar to that used at the BNPP;
2. If possible, tools, equipment and supporting systems are designed similar to their analogues in the "BNPP" work project;
3. Provision is made to prepare the AMB SNF located both at Mayak and BNPP for reprocessing at the DCF.

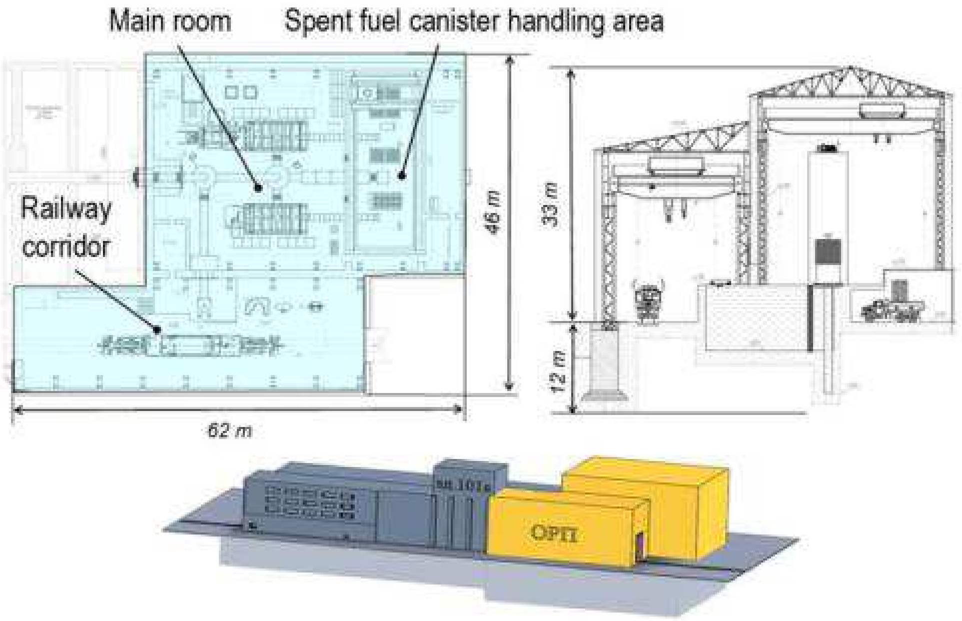


FIG. 5. Layout and configuration of the DCF building at FSUE PA "Mayak".

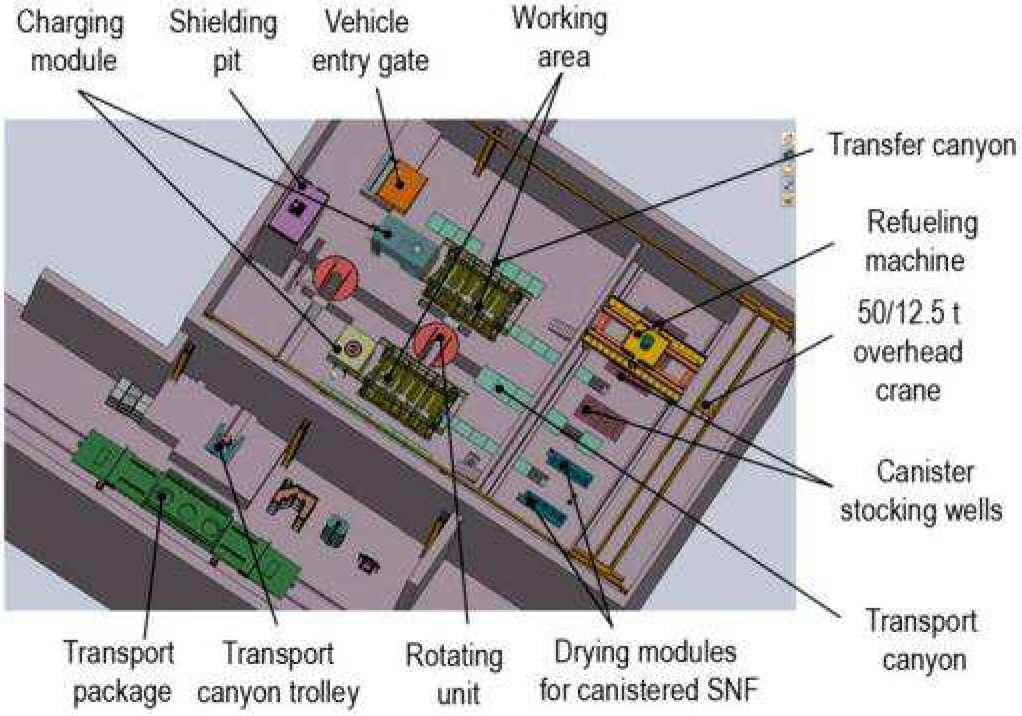


FIG. 6. Arrangement of the basic equipment in the DCF building.

6. CONCLUSIONS

The paper provides the decision on the AMB SNF handling, which consists in the following:

(1) Development of the chemical reprocessing technology for all types of the AMB fuel;
(2) Construction of the facility for cutting the AMB SNF cartridges (DCF) at Mayak and loading of the fuel composition into the tight canisters with the subsequent reprocessing at Mayak site;
(3) Development of the equipment for cutting the cartridges containing failed SNF into fragments, loading of fragments with the fuel composition into canisters and their delivery to PA “Mayak” for reprocessing;
(4) Delivery of the assemblies containing undamaged fuel assemblies from the Beloyarsk NPP to Mayak (DCF), their cutting into fragments, loading into canisters and reprocessing;
(5) The DCF will be used for handling the AMB SNF cartridges located at Mayak.

ORGANIZATION OF WORK WITH ACCUMULATED DAMAGED SPENT NUCLEAR FUEL

V. SMIRNOV
SOSNY
Moscow
Russian Federation

Abstract

Handling damaged SNF requires specific procedures, related to general equipment and technology, but also adapted organization of work, in order to limit radiation issues.

At present manufacturers and consumers of nuclear fuel declare that the probability of nuclear fuel damage in the process of its operation should be no higher than 10^{-6}. It means that during the operating time of one NPP unit no more than one FA and, even, no more than one fuel element will be damaged. Moreover, improvement of nuclear fuel fabrication and operation technologies, application of new safer systems for nuclear reactor operation monitoring contribute to activity that allows complete elimination of fuel damage during the whole NPP operating time. It has been already proven by operation of many NPPs where, though non-upgraded fuel and control systems were used, no damage of FAs located in the reactor core was revealed.

However upon the whole, there were cases of fuel damage as well as incidents and even accidents at the operating NPPs that resulted in generation of significant amount of defective SNF. To a certain moment in the past, SNF handling technologies have not been paid proper attention to as compared with fuel fabrication technologies and fuel operation in reactor cores and it led to and, apparently, will lead to fuel damage. Besides, damaged SNF was also generated for different reasons in other reactors, different from NPPs, such as, for instance, research and naval propulsion reactors. Some damaged SNF was reprocessed, however, considerable amount of SNF is stored in interim repositories and sometimes requires both improvement of storage conditions and resolution on either its reprocessing or final disposal.

To resolve these problems, it is seldom possible to use equipment, technologies and general procedure of work typical for undamaged SNF handling. Besides, in each case there are peculiarities that must be taken into account while designing facilities and equipment and planning work. Sometimes, besides the abovementioned improvements and upgrades, special legal basis is required for activities with SNF, which is elaborated for a single or several projects.

Each case of damaged SNF handling should be treated individually and requires analysis of the following technical issues:

- Evaluation of SNF condition and prevention of its further degradation;
- Justification of safety of SNF handling technologies, as well as technologies for handling of radioactive waste resulting from technological processes, which is an equally difficult task as the first one;
- Separation of damaged SNF from undamaged one, which is extremely important in case of large amounts of SFA.

Handling of damaged SNF also requires proper organization of work, elaboration of relative technical documentation which will help to optimize technological processes and minimize personnel exposure to radiation.

Analysis and due regard of public opinion concerning the problem of damaged SNF handling may add to success of work implementation in each individual case.

Despite a great number of problems, though different in each case, it is still possible, as a rule, to find an adequate solution.