IAEA-TECDOC-1633

Decommissioning of Fast Reactors after Sodium Draining

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IAEA-TECDOC-1633

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INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2009

Sales and Promotion, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 2600 29302
tel.: +43 1 2600 22417
email: sales.publications@iaea.org
http://www.iaea.org/books

For further information on this publication, please contact:

Nuclear Power Technology Development Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria

DECOMMISSIONING OF FAST REACTORS AFTER SODIUM DRAINING
IAEA, VIENNA, 2009
IAEA-TECDOC-1633
ISBN 978–92–0–111809–7
ISSN 1011–4289

Printed by the IAEA in Austria
November 2009

FOREWORD

Acknowledging the importance of passing on knowledge and experience, as well mentoring the next generation of scientists and engineers, and in response to expressed needs by Member States, the IAEA has undertaken concrete steps towards the implementation of a fast reactor data retrieval and knowledge preservation initiative. Decommissioning of fast reactors and other sodium bearing facilities is a domain in which considerable experience has been accumulated. Within the framework and drawing on the wide expertise of the Technical Working Group on Fast Reactors (TWG-FR), the IAEA has initiated activities aiming at preserving the feedback (lessons learned) from this experience and condensing those to technical recommendations on fast reactor design features that would ease their decommissioning. Following a recommendation by the TWG-FR, the IAEA had convened a topical Technical Meeting (TM) on ‘Operational and Decommissioning Experience with Fast Reactors’, hosted by CEA, Centre d’Etudes de Cadarache, France, from 11 to 15 March 2002 (IAEA-TECDOC-1405). The participants in that TM exchanged detailed technical information on fast reactor operation and decommissioning experience with various sodium cooled fast reactors, and, in particular, reviewed the status of the various decommissioning programmes. The TM concluded that the decommissioning of fast reactors to reach safe enclosure presented no major difficulties, and that this had been accomplished mainly through judicious adaptation of processes and procedures implemented during the reactor operation phase, and the development of safe sodium waste treatment processes. However, the TM also concluded that, on the path to achieving total dismantling, challenges remain with regard to the decommissioning of components after sodium draining, and suggested that a follow-on TM be convened, that would provide a forum for in-depth scientific and technical exchange on this topic. This publication constitutes the Proceedings of this follow-up TM held in Cadarache, France, 26–30 September 2005.

The IAEA would like to express its appreciation to all the participants, authors of papers, chairpersons, and to the hosts at CEA Cadarache. The IAEA officer responsible for this publication was Young-In Kim of the Division of Nuclear Power.

EDITORIAL NOTE

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SESSION 2: Reactor and facility decommissioning experimental feedback

SESSION 3: SFR waste characterization and treatment

SESSION 4: Reactor decommissioning strategy

SUMMARY

Background

Since the early 1960s, several countries have undertaken important fast breeder reactor development programs. Fast test reactors were constructed and successfully operated in a number of countries, including Rapsodie (France), KNK-II (Germany), FBTR (India), JOYO (Japan), DFR (UK), BR-10, BOR-60 (Russian Federation), and EBR-II, Fermi, FFTF (USA). This was followed by commercial size prototypes [Phénix, Superphénix (France), SNR-300 (Germany), MONJU (Japan), PFR (UK), BN-350 (Kazakhstan), BN-600 (Russian Federation)], either just under construction, coming on line, or experiencing long term operation. However, from the 1980s onward, and mostly for economical and political reasons, fast reactor development in general began to decline. By 1994, in the USA, the Clinch River Breeder Reactor (CRBR) had been cancelled, and the two fast reactor test facilities, FFTF and EBR-II had been shut down – EBR-II permanently, and FFTF has been until recently in standby condition, but now it is also facing permanent closure. Thus, in the U.S., effort essentially disappeared for fast breeder reactor development. Similarly, programmes in other nations were terminated or substantially reduced. In France, Superphénix was shut down at the end of 1998; SNR-300 in Germany was completed but not taken into operation, and KNK-II was permanently shut down in 1991 (after 17 years of operation) and is now fully dismantled. In the UK, PFR was shut down in 1994, and in Kazakhstan, BN-350 was shut down in 1998.

One consequence of the slowdown in fast reactor technology development programmes is that both active sodium test loops and sodium cooled reactors are currently facing dismantlement. Hence, considerable knowledge and experience is currently being accumulated in the field of decommissioning of reactors and other sodium facilities. Following a recommendation by IAEA’s Technical Working Group on Fast Reactors (TWG-FR), the IAEA had convened a topical Technical Meeting (TM) on ‘Operational and Decommissioning Experience with Fast Reactors’, hosted by CEA, Centre d’Etudes de Cadarache, France, from 11 to 15 March 2002. The participants in that TM exchanged detailed technical information on fast reactor operation and decommissioning experience with various sodium cooled fast reactors, and, in particular, reviewed the status of the various decommissioning programmes. The TM concluded that the decommissioning of fast reactors to reach safe enclosure presented no major difficulties, and that this had been accomplished mainly through judicious adaptation of processes and procedures implemented during the reactor operation phase, and the development of safe sodium waste treatment processes. However, the TM also concluded that, on the path to achieving total dismantling, challenges remain with regard to the decommissioning of components after sodium draining, and suggested that the IAEA convene a follow-on TM that would provide a forum for in-depth scientific and technical exchange on this topic.

Objectives and scope

The objective of the TM was to provide a forum for in-depth scientific and technical exchange on topics related to the decommissioning experience with fast reactors, in particular with regard to the decommissioning of components after sodium draining. Accordingly, the scope of the meeting covers the review and analyses of the experience gained from the decommissioning of both active sodium loops and sodium cooled fast reactors (e.g. KNK-II, Superphénix, RAPSODIE, EBR-II, FERMI, BN-350, BR-10). It is expected that the outcome of the meeting will contribute to the IAEA initiative to preserve fast reactor data and knowledge.

The main focus of the TM was on the decommissioning of both active loop and reactor components (e.g. the primary vessel of a sodium cooled reactor) that have been drained of sodium, but that still conserve some residual amounts of sodium (e.g. films covering the entire surface of the component, or particular sodium heels that cannot be drained).

The TM offered extensive information exchange and discussions on four areas, *viz*. decommissioning and treatment process development (Session 1), reactor and facility decommissioning experimental feedback (Session 2), sodium cooled fast reactor waste characterization and treatment (Session 3), and reactor decommissioning strategy (Session 4).

Session 1: Decommissioning and treatment process development

The Session had four papers: one on the development of the sodium residues neutralization technology by gaseous dinitrogen oxide in cold traps developed by IPPE for the decommissioning of the BR-10 experimental fast reactor, and three on KNK-II decommissioning.

The IPPE method using nitrous oxide (dinitrogen oxide) was employed during BOR-10 decommissioning for the neutralization of non-drained sodium residues. This method is characterized by improved safety, since hydrogen production during the treatment is avoided. It could be applied also to the recovery of residual sodium in tanks and other large volumes.

Three papers presented progress and difficulties in dismantling the experimental reactor KNK-II. The dismantling equipment used for the reactor vessel, as well as various cutting techniques were presented by PROTEM, France. The machining experience and lessons learned when using the PROTEM equipment were presented by Westinghouse, the progress and difficulties in dismantling the reactor vessel by Forschungszentrum Karlsruhe (FZK). An important lesson learned was the need to remove sodium residues prior to dismantling operations. Another important recommendation was not to rely only on intensive testing programs on mock-ups, but to foresee adjustments to the working procedures also during the real dismantling work.

Session 2: Feedback from reactor decommissioning

The experience and feedback from the decommissioning of four sodium cooled fast reactors (Fermi 1, EBR-II, KNK-II and JOYO) was presented in this Session.

Sodium residue cleanup for the Fermi 1 was accomplished with the help of wet vapour nitrogen (WVN) method for processing NaK. Overall, the steam processing experience has been positive. However, lessons learned during sodium processing (e.g. difficult access to work areas, confined work spaces, etc) will prove valuable for future sodium cooled fast reactor designs and decommissioning plans.

EBR-II was shut down in 1994. After processing of the bulk of the sodium drained from primary and secondary loops, residual sodium was transformed into sodium bicarbonate with the help of moist carbon dioxide (a technique tested at ANL). Remaining sodium treatment with moist carbon dioxide followed by steam and water flush is governed by the Resource Conservation and Recovery Act (RCRA). The RCRA mandate issued in 2002 requests removal of all hazardous materials within 10 years. However, the moist carbon dioxide followed by steam and water flush methods is becoming less and less effective, and detailed studies for alternative treatment processes are planned.

The KNK-II paper described dismantling works for major reactor components (vessel and rotating plugs), as well as sodium residual cleaning processes at each step of the KNK-II decommissioning. There were two major lessons learned during these works: firstly, the recognition of the impact of the underestimation of material embrittlement on the primary vessel dismantling process, and, secondly, the importance of residual sodium on the outer surfaces of the dismantled pieces and in the various gaps.

As part of the Mk-III upgrade program of the JOYO experimental fast reactor (implemented to enhance its irradiation capabilities), a 40% power increase was necessary. This entailed the need to replace the intermediate as well as the dump heat exchangers. During replacement works, fuel and liquid sodium were kept in the reactor vessel. Accordingly, impurity ingress had to be prevented and workers' radiation exposure limited. Many important lessons were learned from the replacement work of large components in sodium under difficult conditions like cramped work spaces, closeness to primary sodium boundary and to fuel assemblies with consequent high radiation dose rates, and radioactive sodium treatment. Successful completion of these works and the lessons learned provide valuable insights for future fast reactor designs, construction and operation.

The Session's overall conclusions, based on the final discussion can be summarized as follows:

- The most important factors determining the location of the sodium bulk draining point are linked to the particular design features of the reactor and its circuits;

- In turn, the residual sodium treatment process to be put in place is mainly determined by how judiciously the right bulk sodium drainage method was chosen and how successfully it was implemented;
- A strong requirement to future fast reactor designs is the incorporation of design measures facilitating sodium drainage and cleaning operations.

Session 3: Sodium cooled fast reactor waste characterization and treatment

Five papers gave a comprehensive overview of sodium waste characterization and treatment in decommissioning works for BN-350 and Superphénix, as well as decommissioning planning for Phénix. The overview included radiological characterization, sodium removal technologies, and reactor block dismantling strategy.

The presentations and discussions can be summarized as follows:

- Radiological characterization of activation and contamination is needed for all decommissioning projects to determine waste disposal means. Calculation and measurements are needed to identify all isotopes;
- Minor incidents linked to temporary sodium storage have occurred in the past; therefore needs for temporary storage of sodium must be planned ahead of time;
- Based on the results of experiments on the behaviour of sodium residues in air, it can be concluded that their storage in air can be performed safely for a limited time period, provided dry air is being used;
- Safe storage has been demonstrated for components containing sodium residues, i.e. safe storage rules have been defined for such components, and the use of vinyl/plastic was minimised;
- For BN-300, safety considerations lead to bi-carbonization as the method chosen for the treatment of sodium residues;
- For Superphénix cold traps, based on testing as well as on safety and efficiency considerations, the hot wet vapour nitrogen (WVN) method was selected as the preferred option for the treatment of sodium residues;
- Superphénix decommissioning plans foresee first removing from the reactor all the removable components. The most radioactive components will be removed underwater to significantly reduce dose rates. In the next step, carbonation carbonation will be performed, followed by water fill. Some sodium is expected to remain after carbonation and react during the water fill;
- Different evaluations lead to different conclusions on the ability of the carbonation process to completely remove sodium residues;
- Different evaluations lead to different conclusions with regard to the safety of wet vapour nitrogen (WVN) and steam processes.

Session 4: Reactor decommissioning strategy

The Session addressed the two strategic aspects linked to decommissioning of sodium cooled fast reactors and other sodium bearing facilities, *viz*. sodium residues treatment, and definition of the overall goal of a decommissioning project. For the former, experience made and plans for future decommissioning were presented for (1) the Dounreay fast reactors [PFR and DFR, where the Wet Vapour Nitrogen (WVN) process, developed in 1994, without draining of the caustic soda was used]; (2) the FFTF, where the reactor vessel was cleaned in 7 hours with super-heated steam; and (3) Phénix, for which the current solution for sodium residues treatment is the CO_2 – passivation method.

The papers presented showed how the various definitions of the goal of fast reactor decommissioning are determined by national licensing requirements and costs. The different approaches discussed were ‘green field’ condition (PFR and DFR), ‘entombment of the reactor building without dismantling of the reactor vessel’ (FFTF), and ‘building free from any contamination or activation’ (Phénix).

In the second part of the Session, the meeting heard a detailed description of the decommissioning strategy developed for the French experimental fast reactor Rapsodie, and a paper on the possible impact of decommissioning experience on the design of GEN IV innovative sodium cooled fast reactors.

Two strategic issues were highlighted in the Session's overall conclusions: first, the importance of a comparative assessment (based on an international effort) of sodium residues treatment methods from licensing, costs, and duration points of view; and, second, the need of establishing a methodology allowing 'feeding' decommissioning experience into innovative sodium cooled fast reactor designs efforts.

Overall conclusions

The topical Technical Meeting produced two sets of conclusions addressing design requirements for future sodium cooled fast reactors that, if met, would simplify and ease their decommissioning. These two sets of requirements are based on the experience with and lessons learned from fast reactor decommissioning, as presented and discussed at the meeting. They address, firstly, the reactor block, and, secondly, the various reactor components, and were developed by two ad-hoc Working Groups.

Conclusions for reactor block design of future sodium cooled fast reactors in view of easing their decommissioning

The scope of this conclusions set includes reactor vessel, plugs, fuel handling mechanisms, and other components within the shield tank. The following list of specific conclusions is given:

- Draining provisions or access to vessel bottom for installation of final bulk sodium draining;
- Better access from bottom and external sides;
- Fuel handling machine with better access to remote areas of the tank;
- Leak tight safety vessel;
- Materials selection to minimize potential for high level activation products;
- Select materials that will not corrode and form crevices;
- Build shielding with removable blocks for ease of removal and do not include hard to dispose of materials in these blocks;
- Access for cleaning and drainage in baffle plate area or other structures at the top of the reactor;
- Simpler geometry to prevent pooling of sodium; eliminate un-drainable areas;
- Improve defueling capability and speed;
- Improve ability to remove high activity components;
- Ability to remotely disassemble and remove components;
- Modular rotating plug to ease removal and disassembly;
- Improve maintainability of reactor block internals; which will also improve ability for disassembly;
- Ability to heat the reactor block to keep sodium liquid during fill, draining, and subsequent decommissioning activities;
- Provision of instrumentation and inspection channels for in-service inspection and for use during ultimate decommissioning;
- Accessible coupons for use as activation samples for radiological characterization as well as for machine tool qualification placed throughout reactor block;
- Maintain primary sodium purification to minimize plugging temperature during decommissioning;
- Design considerations to ease graphite removal, or replacement of graphite as moderator;
- High priority to design considerations for ease of removal of highly activated components;
- Aim for standardized designs (pool or loop?);

- Capture on video the whole construction;
- Keep samples of all construction materials;
- Produce and keep detailed as-built drawings with indication of materials, weights, dimensions, surface areas, etc.

Conclusions for reactor components design of future sodium cooled fast reactors in view of easing their decommissioning

The following list of specific conclusions is given, assuming a 1000 MW(e) plant:

- General
    - Design simple drainage methods for all components (e.g. channels large enough for flow, no captive volumes);
    - If not possible, leave access for post operation penetration of components;
    - Define General accessibility requirements;
    - Recognize the need for compromise between design goals and decommissioning goals;
- IHX
    - Innovative core designs that eliminate intermediate heat exchangers (direct heat transfer from primary sodium to steam generator);
- Valves
    - Innovative valve designs without bellows;
    - Innovative sealing systems that eliminate cleaning and residues removal problems;
    - R&D to develop the diaphragm valve (temperature and material related problems);
- Pumps
    - Innovative core designs that eliminate pumps in the primary and/or secondary loop (natural convection core designs)
- Cold traps
    - Innovative cold trap designs with removable, replaceable and washable cartridge
- NaK components;
    - Avoid NaK systems (additional hazards outweigh the benefits);
- Piping
    - Accessible location, minimum bends, no low point traps, minimize imbeds;
- Steam generators
    - Consider once through forced flow (OTFF) designs.

DECOMMISSIONING AND TREATMENT PROCESS DEVELOPMENT

(Session 1)

DEVELOPMENT OF TECHNOLOGY OF NEUTRALIZATION OF SODIUM RESIDUES BY GASEOUS DINITROGEN OXIDE IN THE COLD TRAP OF THE BR-10 RESEARCH REACTOR ON LUIZA TEST FACILITY

V.B. SMYKOV, YU.E. BAGDASAROV, E.A. KRYUTCHKOV, V.S. BELINSKY
STATE SCIENTIFIC CENTRE RUSSIAN FEDERATION (SSC RF) – IPPE,
OBNINSK, RUSSIAN FEDERATION

Abstract

For the purpose of carrying out work on decommissioning of the BR-10 research fast reactor, which is out of operation since 2002, method of neutralization of non-drained residues of alkali liquid metal coolants (Na and NaK) by gaseous nitrous dinitrogen oxide (N_2O) was developed at the SSC RF-IPPE.

1. THE METHOD

The method used is characterized by improved safety because there is no H_2 generation in the course of alkali metal interaction with N_2O. On the stage of experimental studies on this method, mechanism of interaction of Na, Cs and NaK eutectic with N_2O gas was identified. It was shown that hyponitrite of alkali metal (Me) $Me_2N_2O_2$ was an intermediate material assuring complete consumption of the alkali metal in reaction with N_2O. Initial conditions of interaction and temperature effects were determined, as well as composition of the end product (mixtures of oxide, nitrite and nitrate of alkali metal in various ratios) depending on duration of neutralization process, kinetics of N_2O absorption by the alkali metal and gas composition were studied, etc.

Then it were carry out first tests of N_2O-method at examples of neutralization of NaK residues in transport's vessel and in spent hot trap of oxides (HTO) which was cut out from NaK circuit of experimental rig. The results of these tests were positive – all NaK residues in vessel (approx. 1.5 L) and in HTO (approx. 2.0 L) were neutralized and converted in mixture of salts and oxides (KNO_2, KNO_3, K_2O_2, $NaNO_2$, $NaNO_3$, Na_2O) by gaseous N_2O. Then these substances were safely dissolved by water (without release of H_2) and pumped out from vessel and HTO. In this way the equipment were changed-over in safe condition.

Unlike NaK, the Na does not produce the peroxides (Na_2O_2) during interaction with N_2O. But the temperature of start interaction Na with N_2O, unlike NaK ($T_{\text{start interaction}}$ is 30-40°C), is more high – 220-230°C and reaction becomes fast at 250-260°C. So the technology of Na neutralization differs from NaK one.

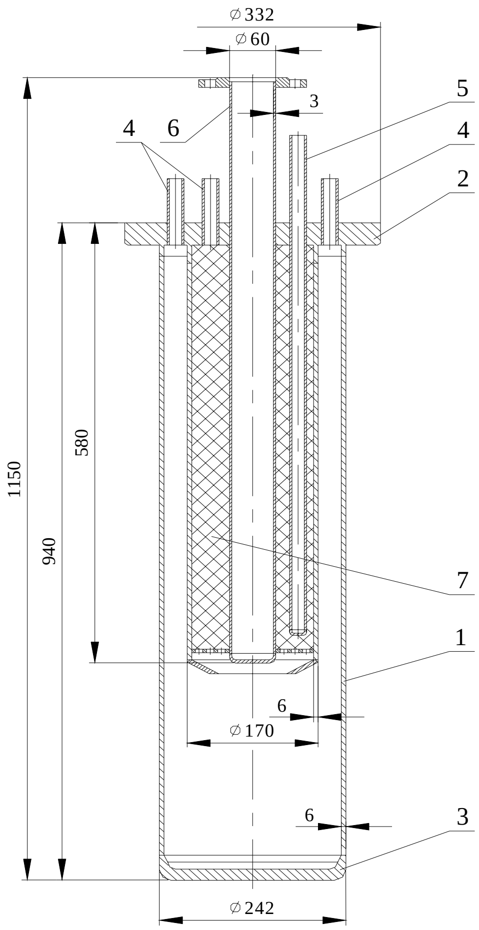
In order to develop technical modes of Na residues neutralization in BR-10 equipment the LUIZA facility was created and put into practice. The facility is fitted out with necessary technical, electrical and process monitoring equipment. As distinct from autoclaves, the facility permits to investigate the Na (or NaK) interaction with N_2O in closed circuit and control conditions during gas pumping.

Figure 1 shows scheme of cold trap from the experimental facility of BR-10 (without external shell).

At the first stage there were trials of neutralization Na residues (each of 300 g) in plain-tube coil. There were carried out 3 tests by different manners of N_2O filling:

1) Periodical inputs of 100% N_2O in tube after discharge the spent gas (N_2) and vacuum treatments, then steady state hold during heating up to T = 250-260°C with measurements of gas composition;
2) Once-through flow of 100% N_2O above (over) Na layer in tube at 250-260°C and constant discharge the gas mixture in atmosphere;
3) The forced closed circulation of gas by pump through coil above Na layer. First and others fills were the 50%/50% gas mixture (Ar/N_2O).

All tests gave full oxidation of initial Na without H_2 release. Final products are N_2 and porous mixture of Na_2O and salts $NaNO_2$ and $NaNO_3$. Process time duration was 16-20 hours. There was no Na_2O_2 in final solid product. The speed of Na layer oxidation in tube was approximately 1.5-2.5 mm/hour.



1 – body of cold trap; 2 – cover; 3 – bottom; 4 – branch pipe; 5 – thermocouple jacket;
6 – housing for heater; 7 – filter packing

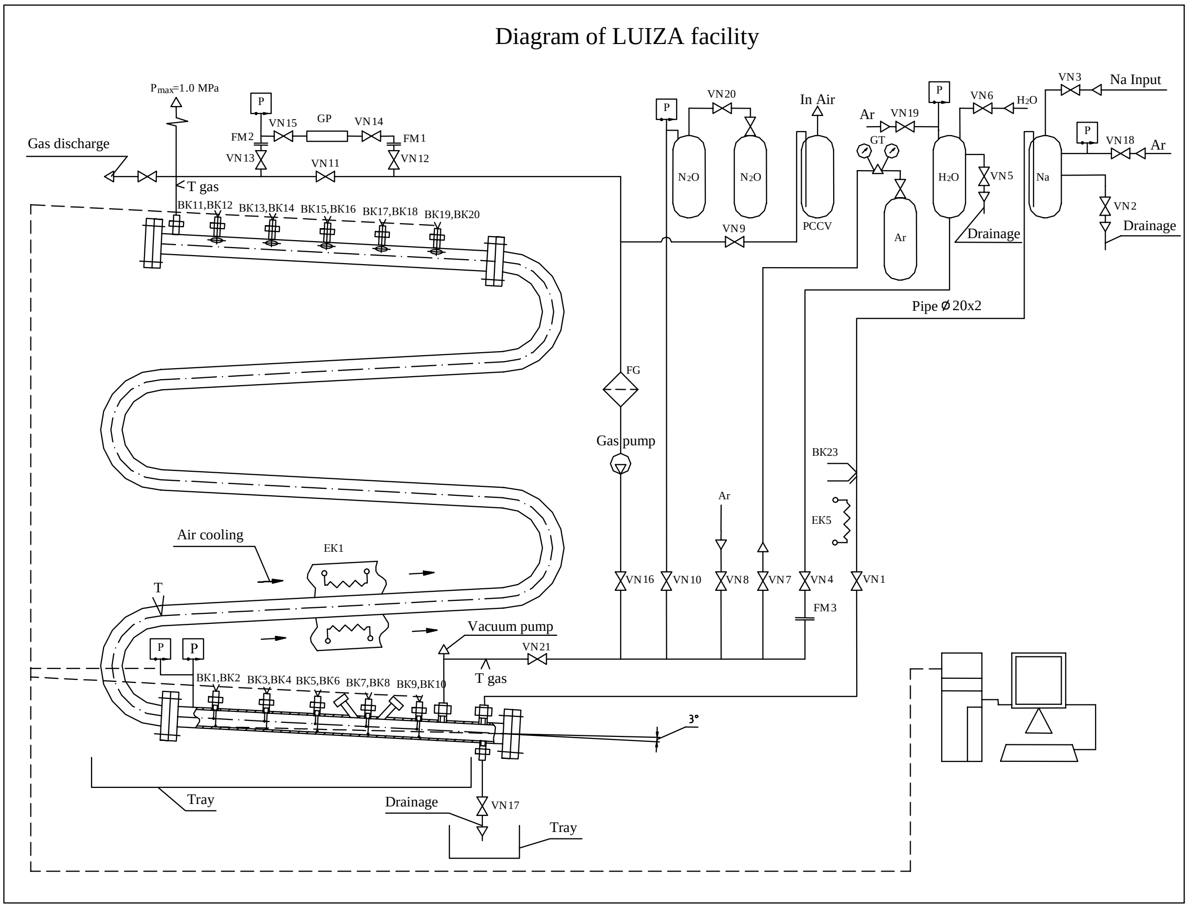
FIG. 1. Scheme of cold trap from the experimental facility of BR-10 (without external shell).

In the test 3) it was pointed out that full Na oxidation may be reached in the bottom of deep vertical ends (h = 110 mm). The final composition of product was as follow: $NaNO_2$ 76-85%, $NaNO_3$ 13-18%, Na_2O 2.3-4.3%. This test has demonstrated that full Na may be converted into the salt forms depending on time processing.

The reaction of Na neutralization by N_2O in tube was calm and safe without of pressure rise. The temperature rises of Na took place only during first 5-7 inputs of N_2O and they were localized in upper Na layer: 200-250°C with input of 100% N_2O; less then 100°C with input of gas mixture 50%/50% Ar/N_2O – they were originated by N_2 molecule formation.

At the final step of process it is needed to rise the temperature up to 300-350°C in order to oxidize the possible Na remainders at the bottom of final mixture and convert the juvenile-formed Na_2O to $Na_2N_2O_2$ and then to $NaNO_2$.The most optimal mode of process was forced closed circulation of 50%/50% gas mixture (Ar/N_2O) by pump through tube above Na layer. In this mode the local temperature rises were less, the temperatures of main Na and tube metal are constant. Generally speaking, the mode of N_2O process must be selected with reference to specific conditions of equipment with Na (NaK) residues and chemical state of theirs.

At second stage we have carried out the experimental test of N_2O process at spent cool trap (CT) which has contained the unknown mass of Na residue with NaOH and Na_2O. The initial chemical composition of residue was unknown too. Judge by results which were obtained by electrical level indicator, CT has contained approximately 6-8 L of residue. The free gaseous volume in CT was about 33-34 L. This spent cold trap was installed in LUIZA test facility (Figs 2-4).



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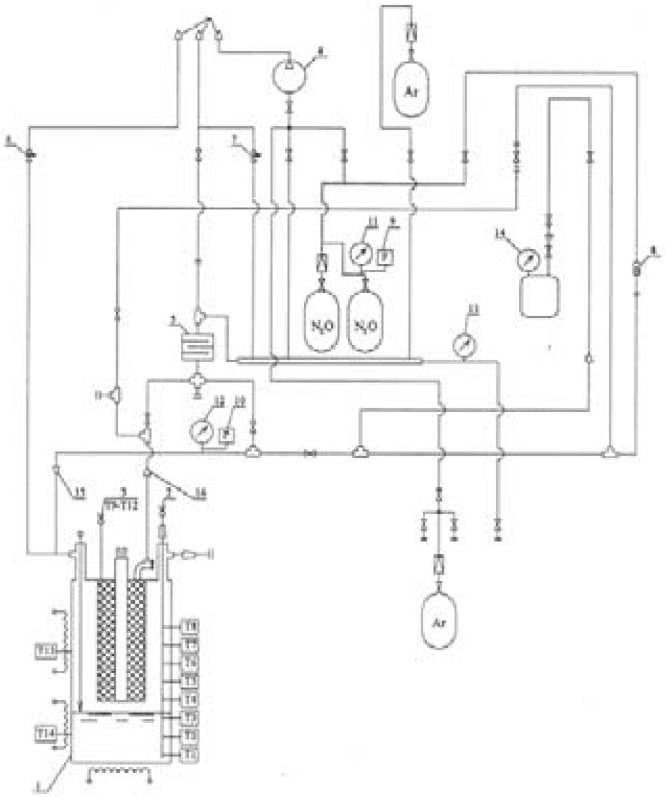
IG. 2. Diagram of LUIZA facility.

FIG. 3. Picture of LUIZA facility.



1 – cold trap
2 – thermocouple probe
3 – thermocouple probe
4 – vacuum pump
5 – dust trap

7 – safety valve
8 - flowmeter
9 -10 pressure transducer Sapfir
11-14 compound pressure and vacuum gauge
15 - feed N_2O
16 - gas discharge
T1-T14 – thermocouples

FIG. 4. Diagram of cold trap in LUIZA facility.

During 45 hours of LUIZA test facility effective operating time, free sodium in the cold trap was neutralized by the dinitrogen oxide into the salt forms. At N_2O process the highest temperature inside the CT was 580°C at first N_2O inlet (we used the 100% N_2O). During next N_2O inlets the temperature rises has became less and less. The N_2O process has accompanied with N_2 release and common gas pressure decrease. We did not use the forced gas circulation because consumption of each portion of N_2O was fast – through several minutes after each N_2O inlet in CT his concentration in gas mixture N_2O +N2 was less 10%. The initial pressure of N_2O in CT was 2.0-2.5 ata (atmosphere absolute); the final gas pressure was 1.1-1.5 ata. At the final step of N_2O process we revealed the full absorption of N_2O without of N_2 release and without temperature rise, besides the common pressure became less then 1 ata. We have tested different modes of N_2O inlet in CT – by portions of 40 L through the filter packing and in settler with time hold, constant gas flow (20 L N_2O/hour) through CT with discharge of gas mixture (N_2>90%, N_2O<10%) in ventilation system. After 45^{th} N_2O input the concentration of N_2O in discharge gas mixture has became more then 10% (up to 50%), after 49^{th} – up to 90%, and N_2O process was stopped. Overall consumption of N_2O was 3115-3120 L for the neutralization of Na residues in this cold trap.

During the whole time of N_2O process we have took the samples of neutralization product from settler of CT. Chemical analysis of last sample has shown that content of free Na in product was 0.03-0.07%(mass.). This value is 10 times lower then criterion value (0.5%) stated out in Program of this study. The main part of last sample was solid mixture NaOH+Na_2O (60-70%). The presence of very mach amount of NaOH did not prevent practically full neutralization of Na residues. Such great content of NaOH in Na residues is not typical for spent reactor's cold traps. It is specificity only for this CT.

In general, reprocessing of non-drained sodium residues containing impurities in the cold trap is easily and effectively controlled by varying supply of nitrous oxide.

The main difficult task was to dissolute the obtained product in settler in water. It was an unexpected problem. The reason was as follow – the sodium salts ($NaNO_2$, $NaNO_3$) formed the easy-fusible eutetic mixtures with NaOH. Their fusion temperature is 230-240°C. During N_2O process these eutectic mixtures had been melted some times and finally gave the solid steady state product resist attack by water. On the contrary, in filter packing the product was easy soluble in water. The cold trap was fully filled by water and hold with water during 100 hours. There was not detected the H_2 release in Ar protective atmosphere, but after 100 hours of exposure in water only 4 kg of product in settler were dissolved in water. The rest product in settler was mechanically extracted from settler after cutting the CT.

The chemical analysis of rest product has shown as follow: 70-74% $NaOH+Na_2O$, 10-15% $NaNO_2+NaNO_3$, 5-7% $Na_2N_2O_2$, 3-5% corrosion products collected in settler of CT. The free metallic Na in product was not detected. The weight of extracted product was about 6 kg. Summary mass of product in settler before water dissolution was about 10 kg. The upper part of product was porous.

The results of experimental test of N_2O process at this spent cold trap have shown that instead of water dissolution of final product (mixture of sodium salts and oxides) the technology of its solidification in spent CT can be developed. Decision was made to carry out trial neutralization run of sodium residues in the cold trap of the secondary circuit of the BR-10 research reactor by N_2O gas.

This work is supported by ISTC Project N 2911.

REFERENCE

[1] BAGDASSAROV, YU.E., SMYKOV, V.B., SARATOVA, I.G. et al., Report N 2911.01.01, ISTC (2005).

DISMANTLING EQUIPMENT AT KNK-II

E. BENSOUSSAN
PROTEM SAS, ZT LES BOSSES, ETOILE SUR RHONE, FRANCE

Abstract

The paper describes the special machining tool used to dismantle reactor and reactor internals of the experimental sodium cooled reactor KNK-II.

1. SYSTEM DESCRIPTION

The Protem machining unit F9-2000 is scheduled to work on the experimental sodium cooled reactor KNK-II at the research center of Karlsruhe. It has been engineered to achieve the complete dismantling of the reactor and of its internals.

The decommissioning is being performed from the top of the vessel till reaching the lower level of the internals. The final cutting of the vessel will than be achieved from the bottom to the upper level.

The complete design of this equipment has been done according to strict specifications, which were mainly dictated by the maximum possible weight of the machine, the sodium environment, the nitrogen atmosphere and the required final shape of the dismantled parts.

When considering all those restrictions, the most proper technical cutting process was mechanical cold and dry cutting.

The final target of the complete operation consists in cutting all the internals and the vessel itself, taking into account all the specifications previously mentioned. Additionally, each of the dismantled parts, have to be sized in order to fit into their drums.

The cutting techniques being used with this equipment are: milling with high speed tools steel bits, circular saw blades, face milling cutter, milling bell, and hydraulic shears.

The equipment (Figs 1 and 2) is composed of a supporting frame made up of a clamping system, the H vertical rotation axis and all the systems for safety manual releasing. This part is set up on the bridge connecting plate, in order to get all the energy supplies which are hydraulic, electric or pneumatic. The main body, is equipped with a V horizontal rotation axis holding a crossed table with X and Z linear axes. The connection between the lower part and the upper part features an additional linear movement named X+ for an easier positioning when cutting the walls of the reactor.

Total cuts to be performed amount up to more than 1300 pieces which could weight up to 300 kilos each.

A specific crane has been designed to hold the machine at the requested working height.

FIG. 1. PROTEM machining unit, detail.

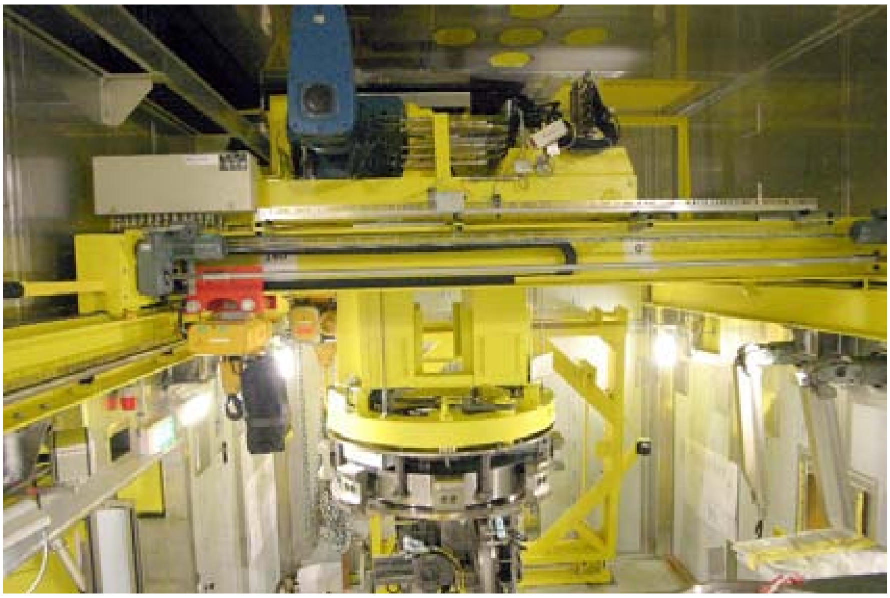


FIG. 2. PROTEM machining unit, general view.

The F9-2000 is built for clamping into the reactor vessel by the means of 8 hydraulic clamping jaws (Fig. 3).

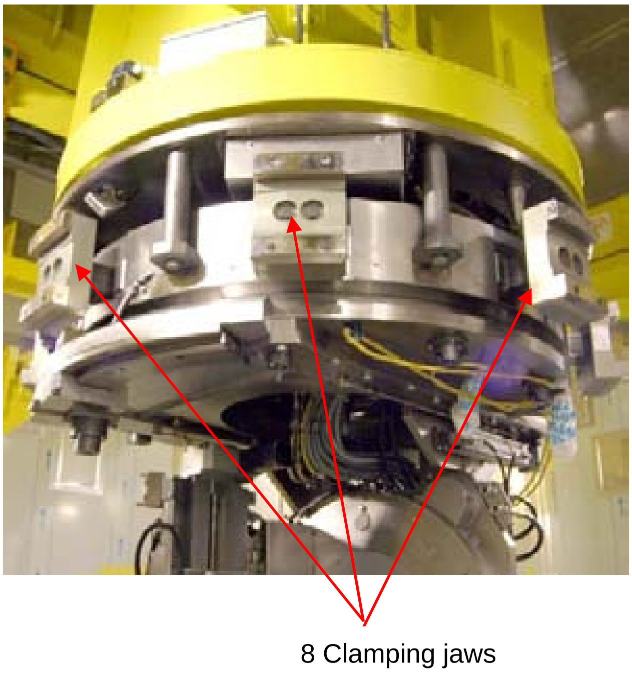


FIG. 3. Clamping system.

The clamping system (Fig. 3) is actuated through a cam with 8 slots enabling all of the 8 clamping jaws to be expanded and retracted simultaneously. The clamping stroke is 81 mm.

- The diameters of clamping according to the clamping jaws are:
  - (i) Type 1: minimum = 1754 mm, maximum = 1916 mm
  - (ii) Type 4: maximum = 1862 mm, minimum = 2024 mm
- Clamping pressure: 140 bar
- Hydraulic cylinders: diameter 63 mm, stroke = 400 mm
- The axial force applied on each clamping pad is 5500 DaN.

This unit is equipped with six axes that are managed through the monitoring control (Fig. 4).

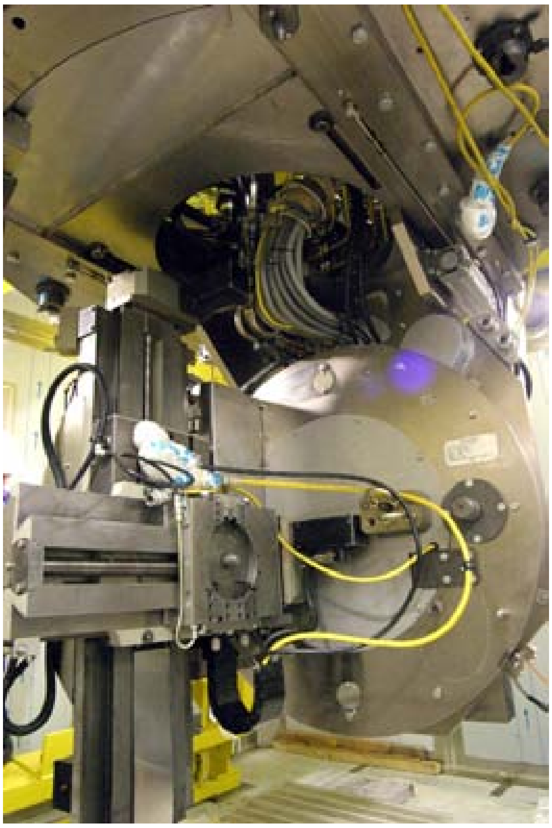


FIG. 4. Clamping system, detail.

The operator can have access to all the data concerning the parameters and the positioning of the entire axis during the operation.

When a specific axis or module is actuated, the live parameters can be read and therefore any information that should draw the attention of the operator is available.

The H axis has a capacity of a 362° rotation and the maximum speed of 0.2 rpm. The axis position is set during a start up procedure.

The V axis has also a capacity of a 362° rotation. Its position is set through an initialization procedure of translation of the Z-axis.

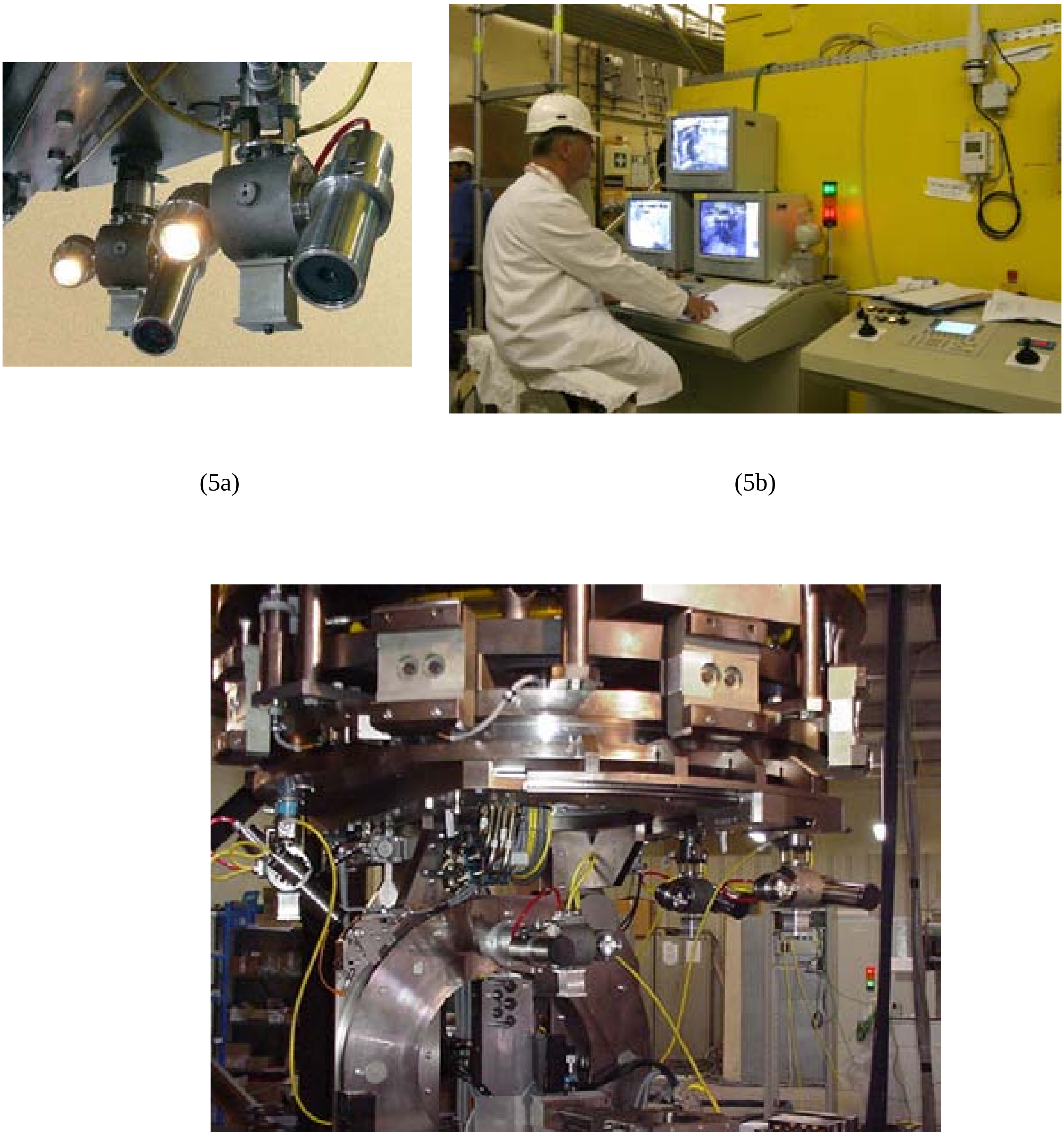
The Z-axis on which is mounted the X-axis has a translation stroke of 700 mm.

The X axis has a translation stroke of 470 mm.

Some specific phases require additional stroke Z+ giving an additional 140 mm and the X+ an additional 240 mm.

The operator keeps always the visual and acoustic access of the working area and its environment through the remote video system.

The dismantling machine F9 2000 features a visual inspection of the decommissioning through a 4 camera remote video system with projectors and a microphone connection. It is a complementary control system comforting the reading of the parameters at the control panel (Fig 5a,b,c).



(5a)

(5b)

(5c)

FIG. 5 a,b,c. Control panel.

All the dismantled parts are being lifted with either the module being used during the cut or with specific magnetic or gripping modules. The secondary waste produced during the cuts is either being sucked through a special tele-operated vacuum system or brought back through magnetic modules. The tests carried out in PROTEM workshop (Fig. 6a,b) have required the achievement of a specific pool for the setting up of a full size mock-up in diameter of the reactor and its components, along with its specific crane. The complete engineering of the dismantling phases could then be tested according to all the studies that were validated.

(6a)

An airtight cell under a nitrogen atmosphere has been erected on site (Fig. 6b) so as to avoid sodium ignition during dismantling operation.

(6b)

FIG. 6a,b. PROTEM equipment during test process.

The dismantling equipment is traveling within the cell with a special engineered crane. The various connecting plates are shown in Figs 7-10.

FIG. 7. Hydraulic connecting plate.

FIG. 8. Electrical connecting plate.

FIG. 9. Centering pin.

FIG. 10. Rotative lock.

The crane is monitored through its own control cabinet (Fig. 11).

FIG. 11. Crane monitoring.

There are 6 modules for machining. All the monitoring is done through the control panel.

All the modules are tele-operated and the hydraulic, electric or pneumatic connections are done through either a light or a heavy-duty manipulator. The following figures provide examples of modules. Figure 12 shows an assembly that is mainly used for performing cuts away from the center and as close as possible to the wall of the reactor vessel.

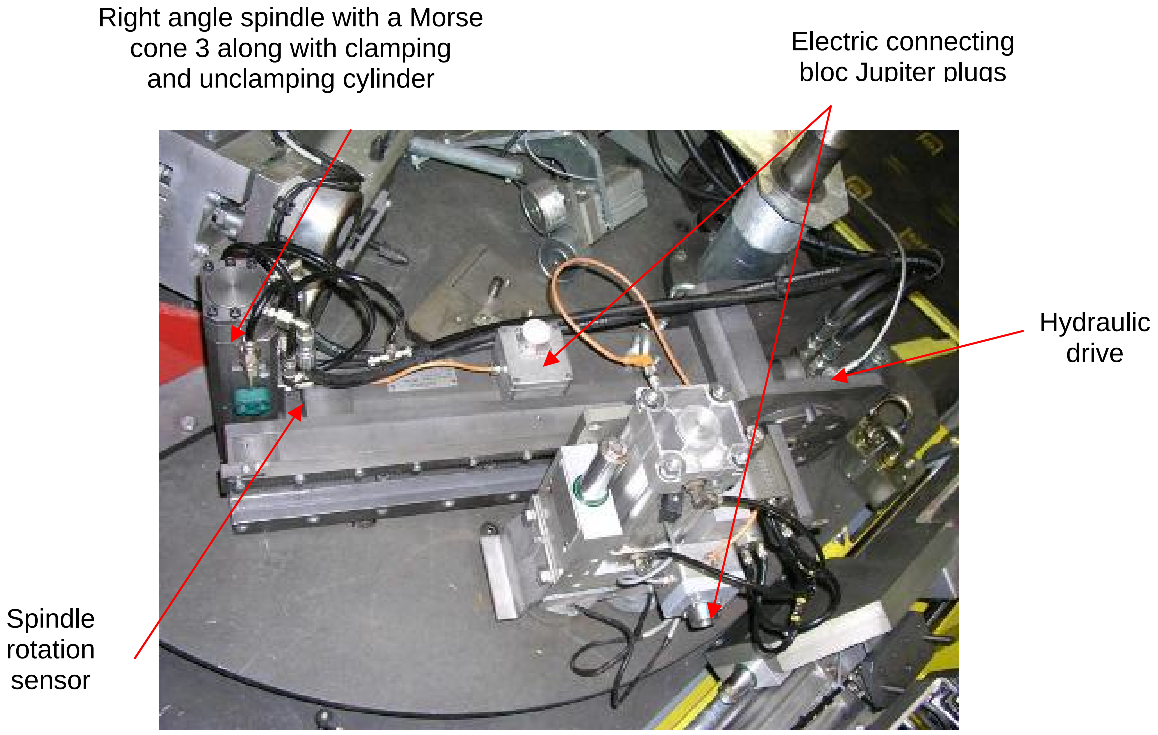


FIG. 12. Cutting module.

The module with shears (Fig. 13) is mainly used to cut piping at the bottom of the vessel, but also to bring back some of the segments.

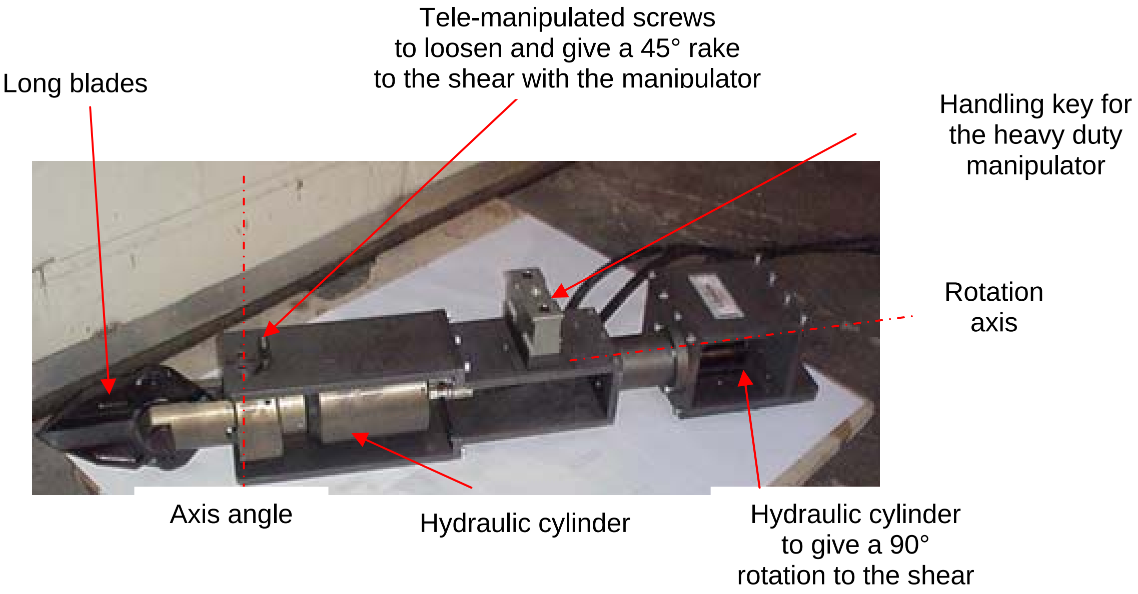


FIG. 13. Module with shears.

The 'Module 12' (Fig. 14) is the main cutting module. It is used at almost all stages of the dismantling process. It can be coupled to other gripping modules.

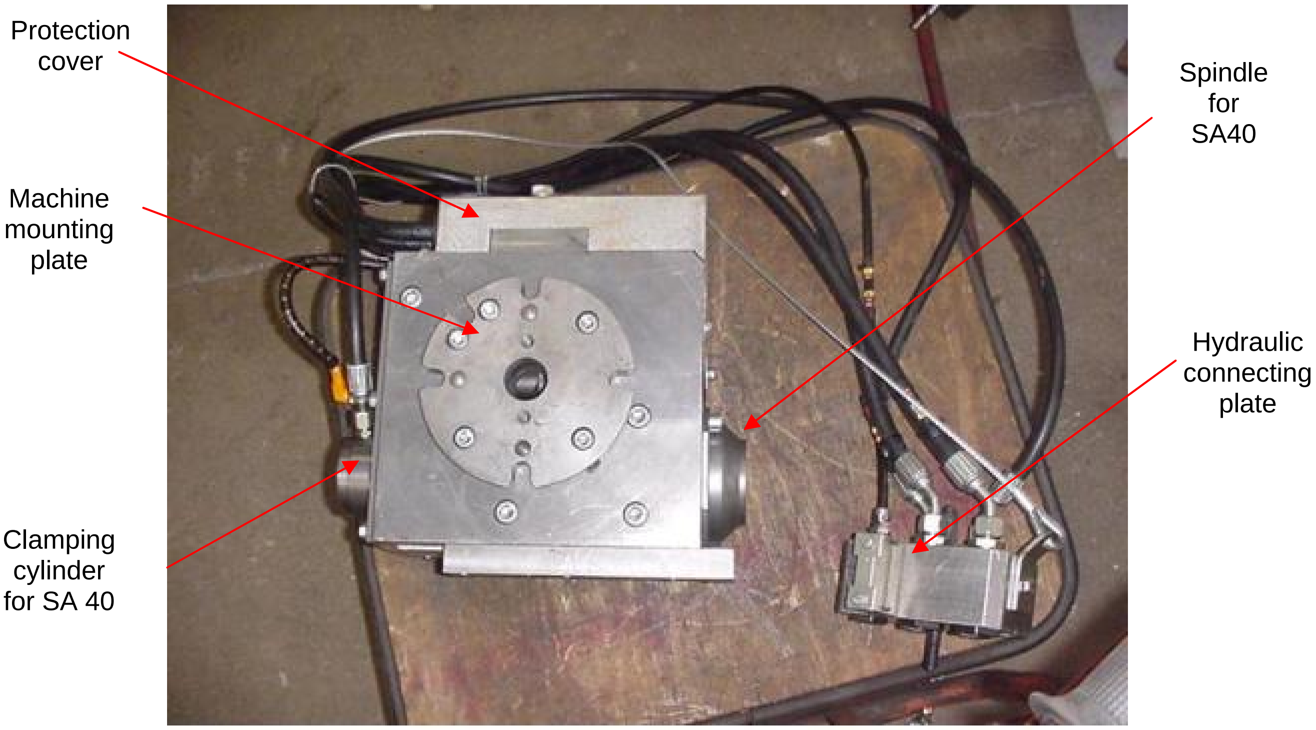


FIG. 14. The 'Module 12'.

Current maintenance of the machine is performed, like for all normal operation, with the help of manipulators.

As all the operations the current maintenance of the machine is done through the manipulators.

The chuck and the different milling cutters are shown in Fig 15 a-d.

(15a) SA 40 chuck.

(15b) Milling bit used for heavy wall.

(15c) Milling cutter 'Bell' type: for the grid plate wall thickness 85 and 120 mm.

(15d) Milling saw cutter with an extended shaft to cut inside connecting tubes.

FIG. 15a-d. The chuck and the different milling cutters.

The ‘Module 15’ (Fig. 16) is used to cut the wall of the vessel (from 6 up to 16 mm wall thickness while generating fewer chips) with 104 mm diameter circular saw blades

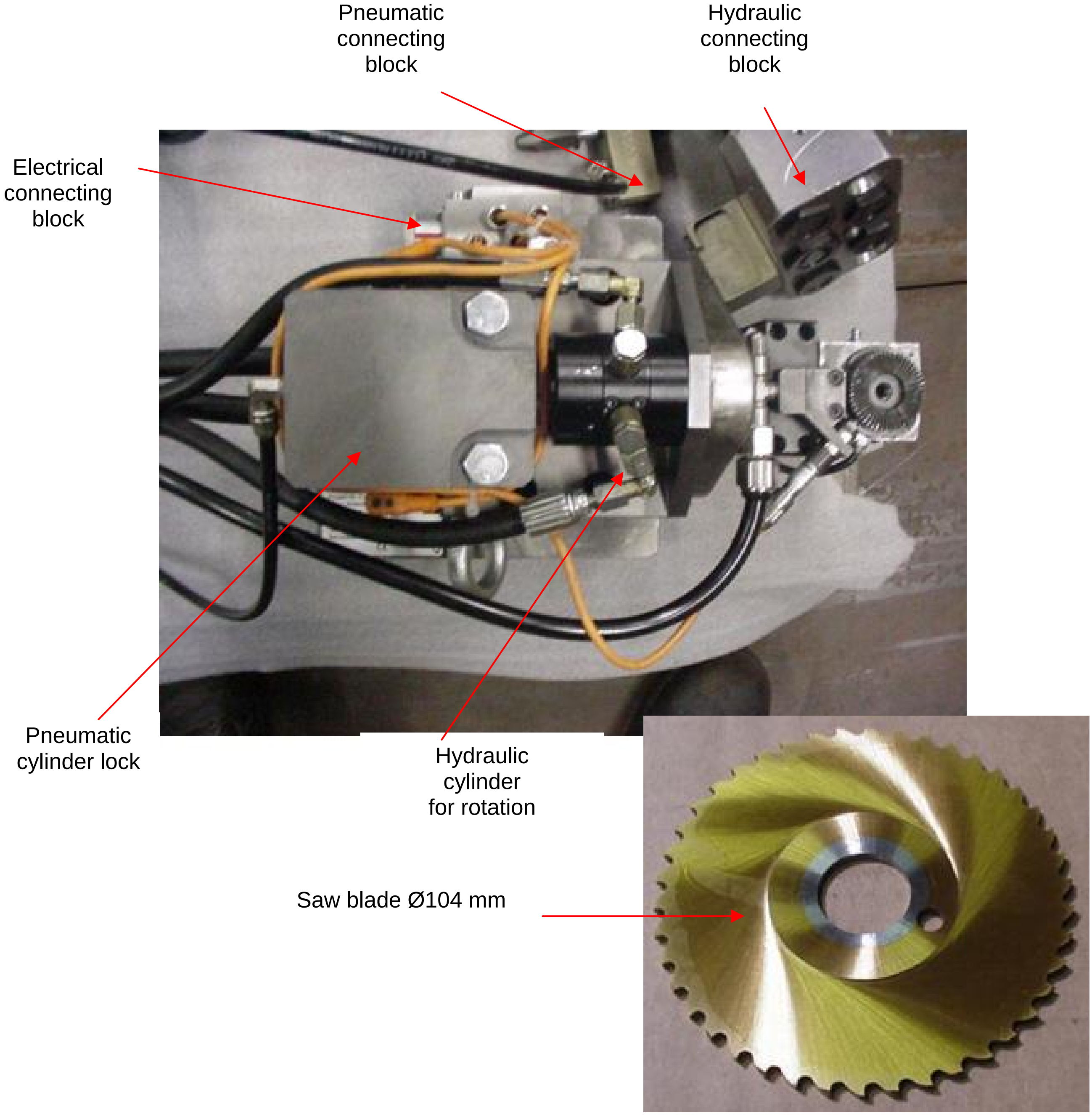


FIG. 16. The ‘Module 15’.

This system is used to absorb the possible vibrations by generating load near from the cutting zone. The load is applied through pneumatic cylinders. There are two anti vibration systems (Figs 17 and 18).

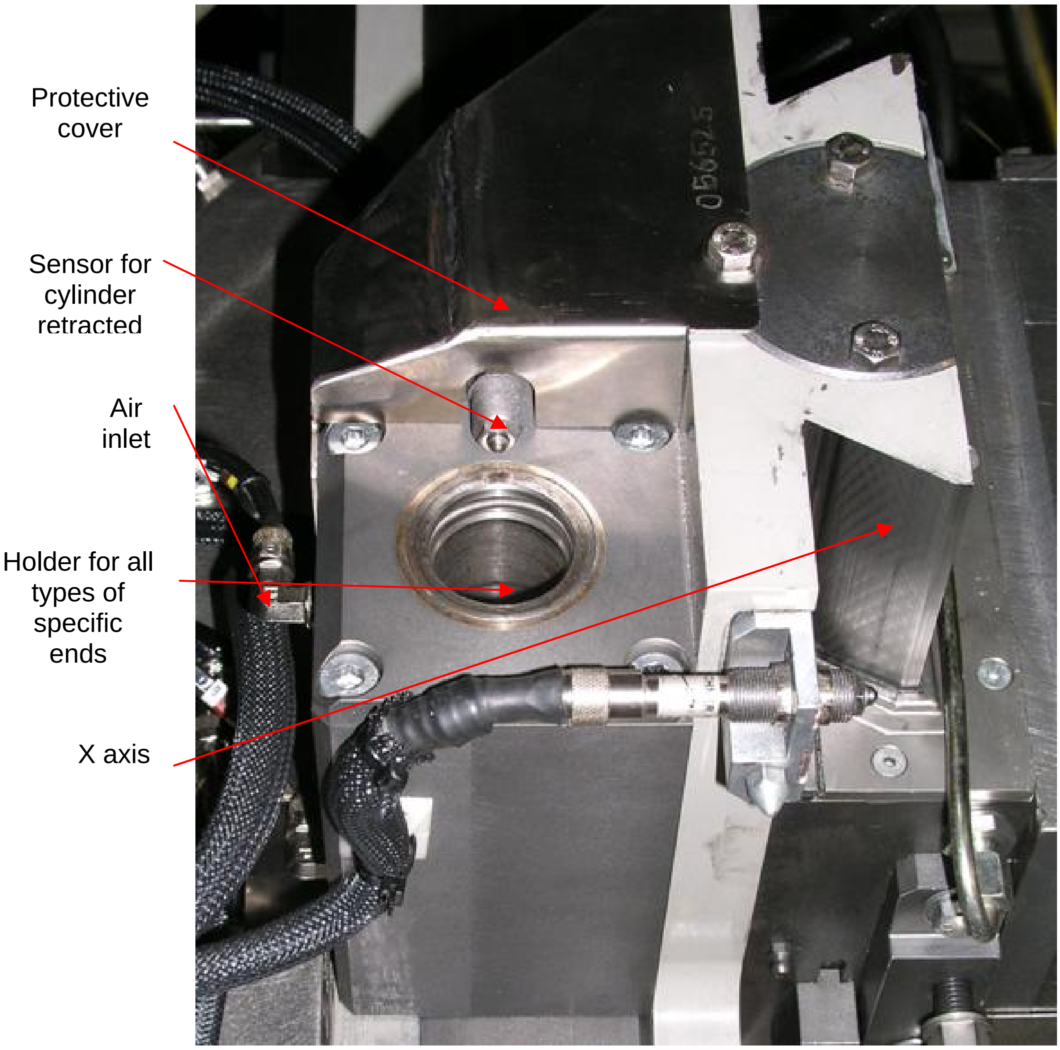


FIG. 17. Anti Vibration System # 1.

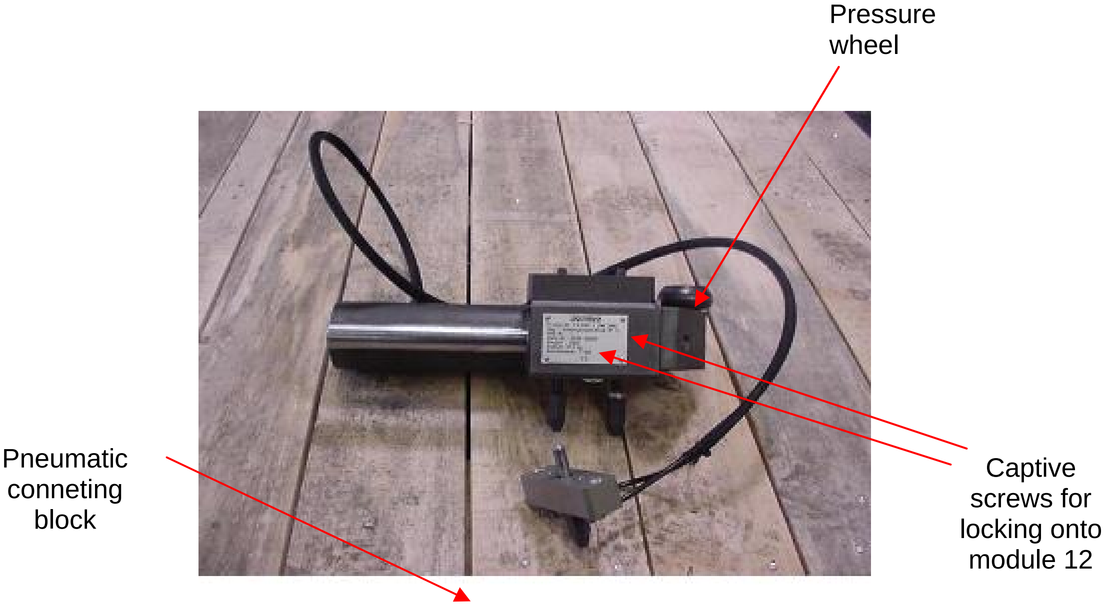


FIG. 18. Anti Vibration System # 2.

The F9 2000 features 9 different modules for holding the ‘wastes’. Figure 19 shows one of the 2 main modules, whose task is to hold the segments cut from the wall of the reactor vessel.

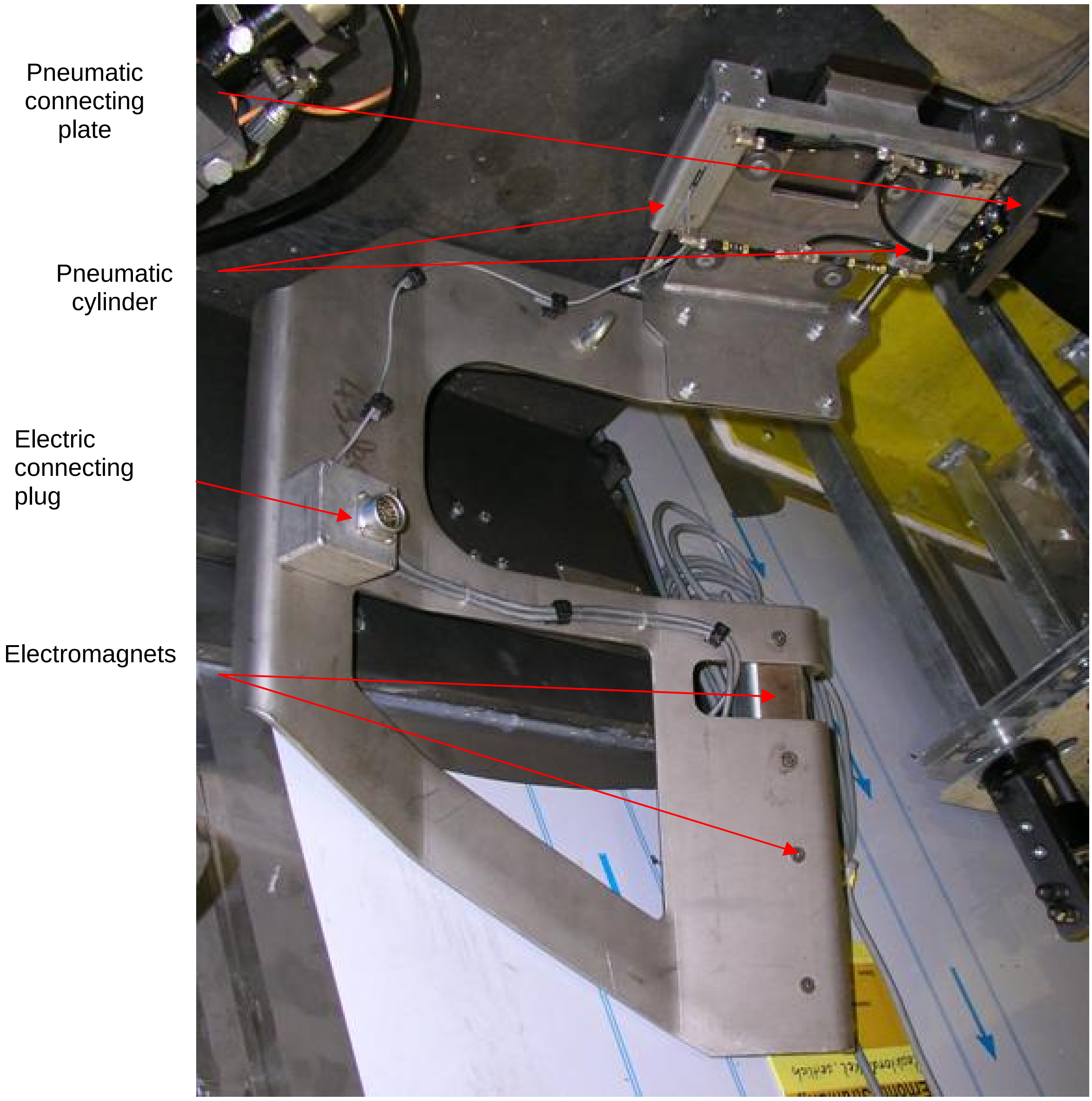


FIG. 19. Main Waste Holding Module.

The Module 11-1 (Fig. 20) is equipped with electromagnets and can be used for holding chips.

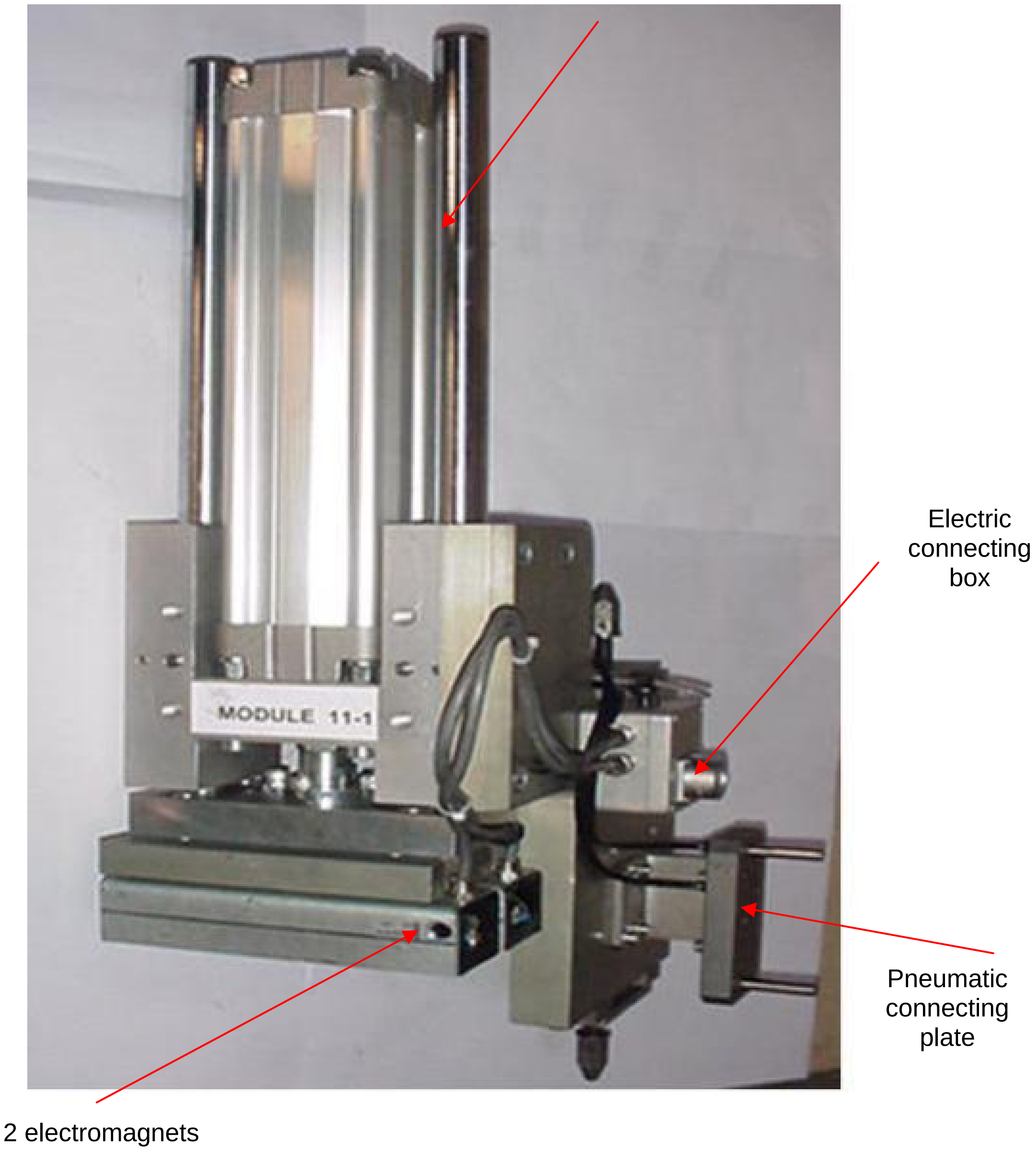


FIG. 20. The Module 11-1.

The Module 11-2 is shown in Fig. 21.

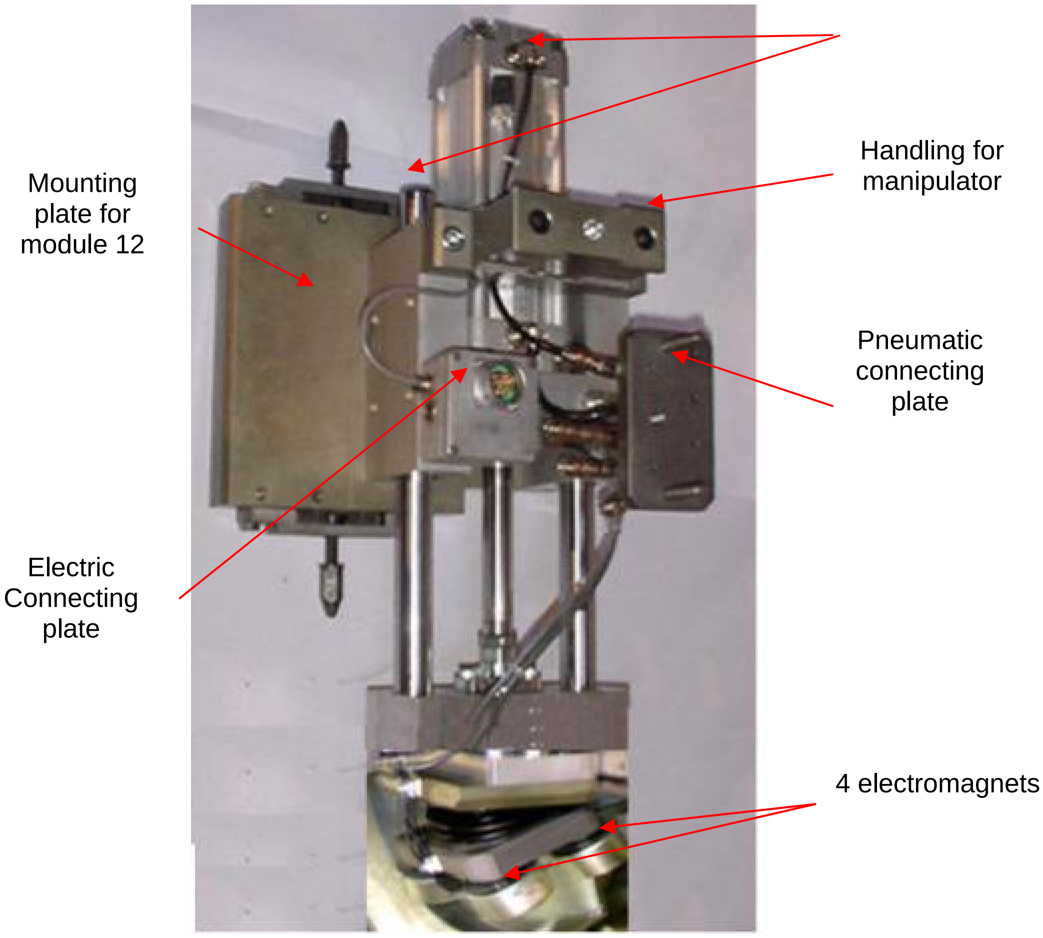


FIG.21. The Module 11-2.

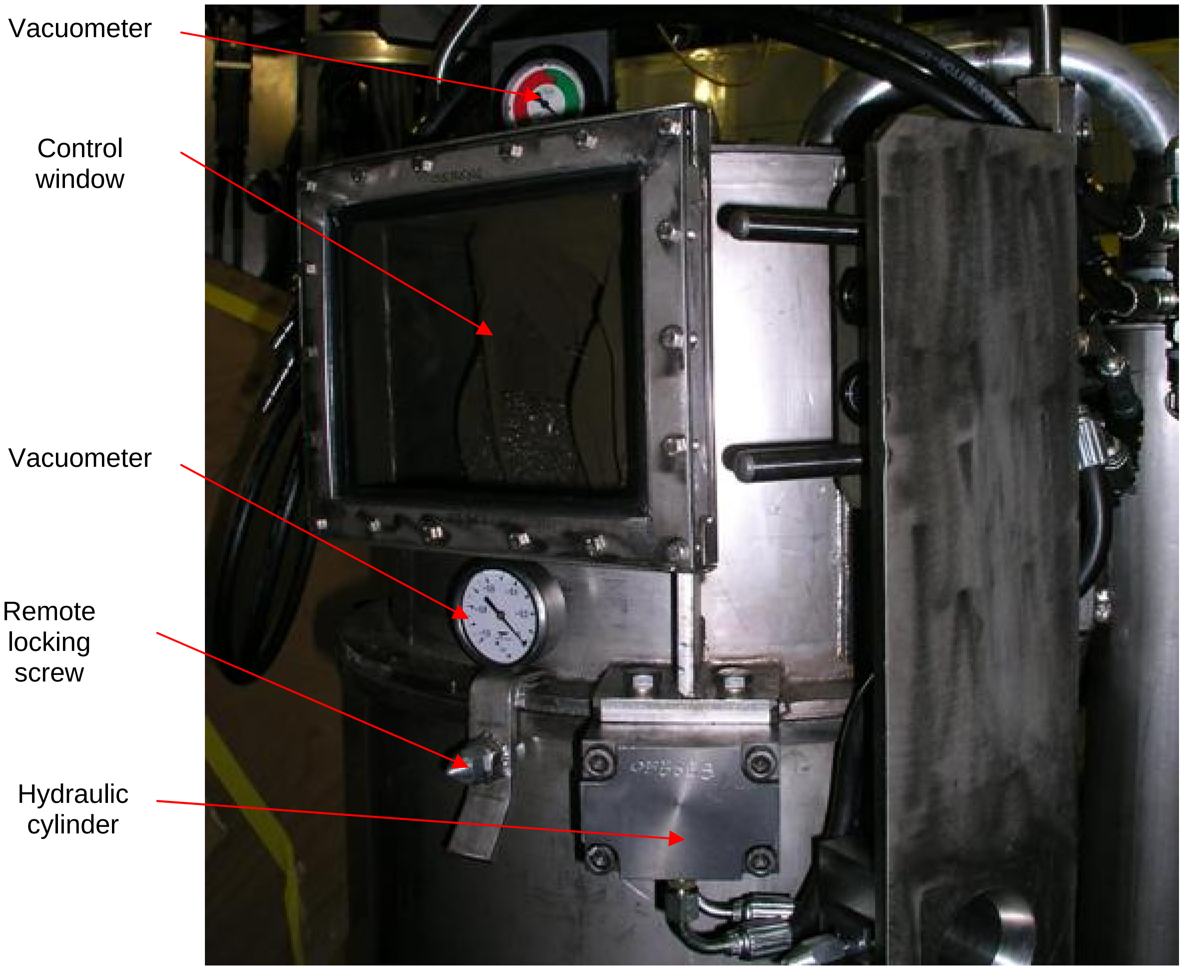
The Module 11-4 (Fig. 22) is used to collect the segments of the reflector and from the thermal shield. It grips parts up to 300 kg.

FIG. 22. The Module 11-4.

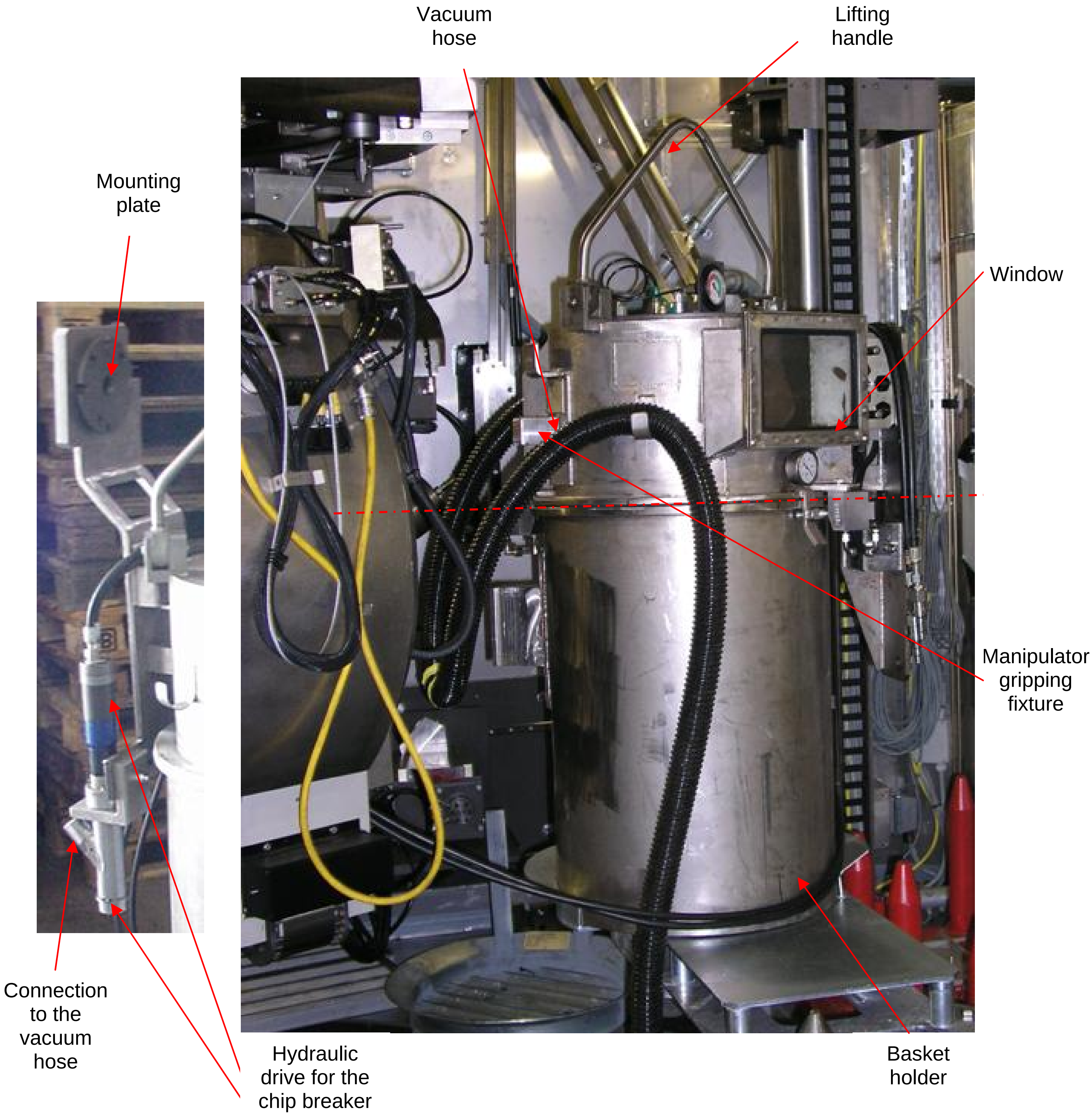
The Module 10 ‘Chips Vacuum Systems (Fig 23a and b) consists of: suction head, basket holder and vacuum arm.

The window allows visual access with the help of the video cameras, enabling the operator to visually control the level of chips collected. With the 2 ‘vacuometers’ the operator can control whether the filter or the suction head are jammed.

While sucking the chips, a specific mechanical motorized chip breaker separates the massive block produced by the accumulation of chips.



(23a)



(23b)

FIG. 23 a,b. Module 10 ‘Chips Vacuum Systems’.

The aim is to collect the chips laying at the bottom of the vessel while avoiding jamming the inlet of the vacuum head.

Tests have proven the efficiency of the system with an average of 1kg of chips sucked per minute.

This assembly (Fig. 24) can hold the modules in order to carry them from the reactor room to the maintenance room. It is mounted onto the main carrier, which is the machine 'bed'. It is fixed with the help of 2 large pins which are also used for the positioning of the dismantling machine. Therefore it can be placed in a reachable area for the manipulator.

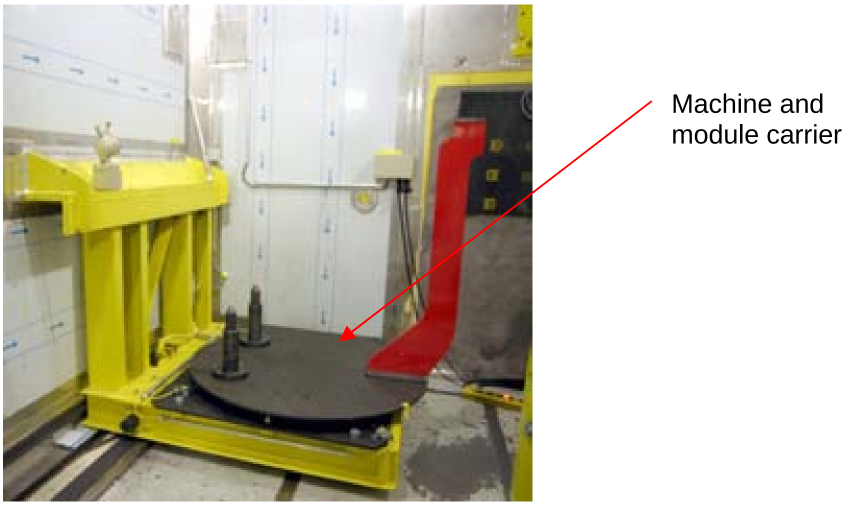


FIG. 24. Machine and module transfer carrier.

This component (Fig. 25) is mounted onto the cell wall. It features a rack mounted with 2 slides for horizontal translation movement. It can be locked with the manipulator. The modules are set on the rack plate.

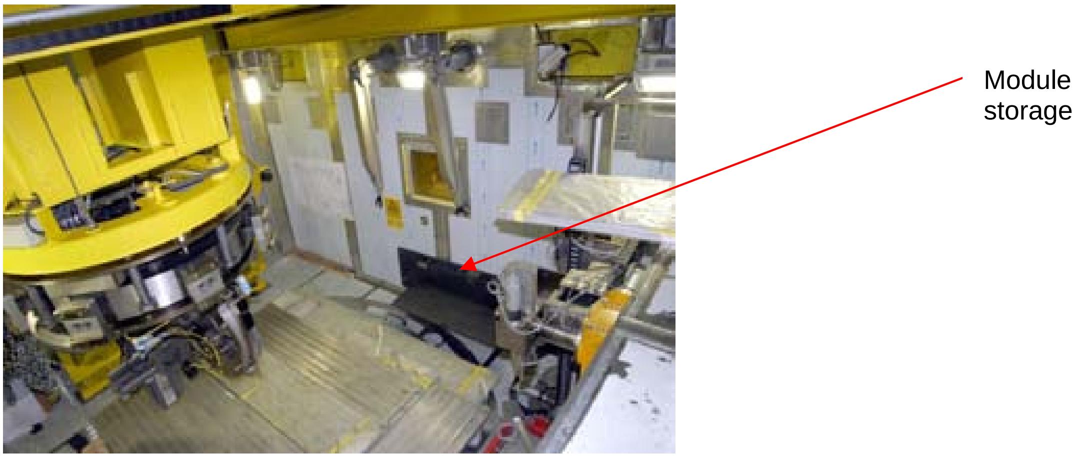


FIG. 25. Module storage.

This platform (Fig. 26) can hold all the tools with or without their holders.

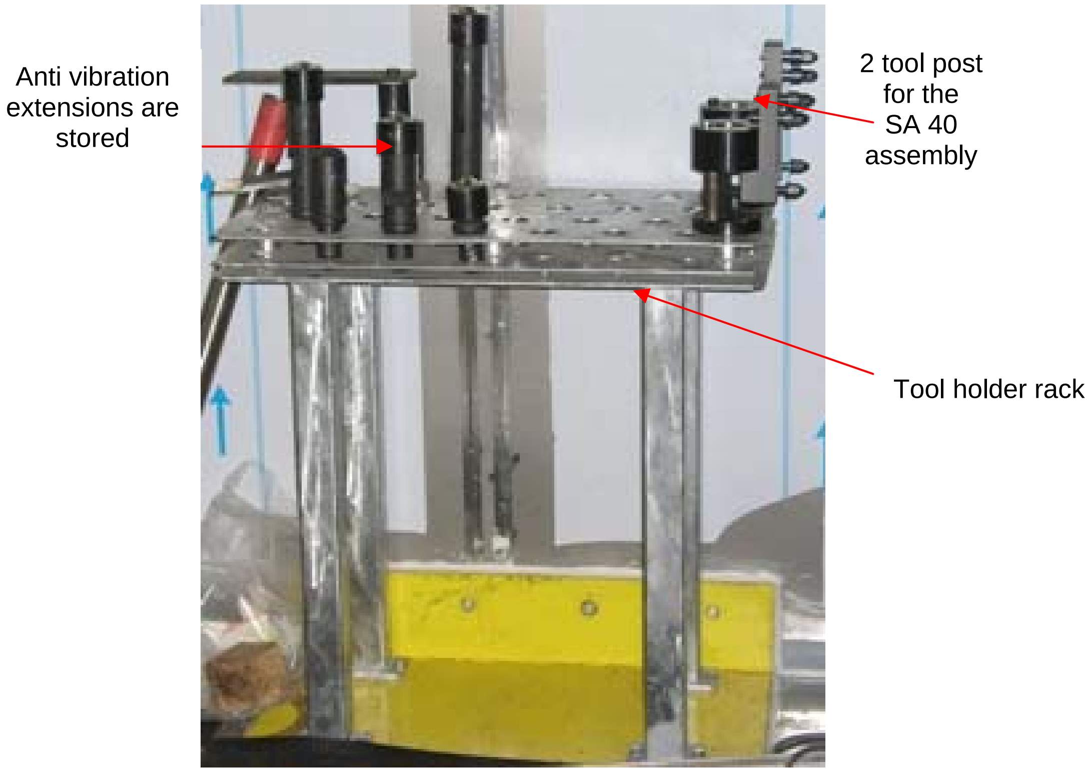


FIG. 26. Tool holder storage.

Through the control panel, the operator has access to the following data: nitrogen pressure, hydraulic pressure, axis intensity, and temperature.

All these information is transmitted to the operator either as written protocols on the control panel, as blinking messages, or as an audible alarm, depending on the level of the emergency.

PROGRESS AND DIFFICULTIES IN DISMANTLING KNK-II REACTOR

J. MINGES, K. BROCKMANN, I. HILLEBRAND, W. PFEIFER
FORSCHUNGSZENTRUM KARLSRUHE GMBH, GERMANY

Abstract

The compact sodium cooled reactor facility KNK was an experimental nuclear power station with 20 MW electrical power output. Initially, between 1971 and 1974, the plant was operated with a thermal core and referred to as KNK-I. Between 1977 and 1991, it operated with a fast core as KNK-II. The reactor is now under decommissioning completely down to green field conditions following ten permission steps according to the German Atomic Law. Current permission step nine includes the dismantling of the reactor vessel with internals and the surrounded shielding. Because of residual sodium contamination within the reactor vessel and the activation of the steel all dismantling work has to be done remote controlled under nitrogen atmosphere. Supported by special equipment within the reactor building, sodium contaminated pieces will be treated and transferred sodium-free to the central waste treatment center (HDB). Since December 2004 all equipment is in operation and the decommissioning of the reactor vessel is under progress. 'green-field' status of the site is planned for 2013.

1. INTRODUCTION

Cause of high activated material and residual sodium an enclosure above the reactor vessel with a special venting system, run with nitrogen atmosphere, within the reactor building was required. The enclosure contains all equipment for:

- Cutting the reactor vessel in pieces;
- Filling pieces in washing baskets;
- Transferring the baskets out of the enclosure without contamination;
- Interventions for repair and service;

2. DESCRIPTION OF OPERATION

Baskets with sodium contaminated pieces are transferred to the sodium washing pit to convert the sodium. Pieces without sodium (cleaned or not sodium contaminated) are transported to the central waste treatment center (HDB) of the research center nearby. Disassembling and washing are separate tasks, in order to work independently from each other. The main tool is the disassembly manipulator. It has 6 axes to make all necessary cuts and handling of the cutted pieces, also the occurred chips. Different modules for cutting and handling the pieces can be adapted. Up to now about 5.5 mg (of 43 mg) are dismantled. Some difficulties at first operation time resulted in a significant time delay. First rank has the changes of steel material properties. On the other hand a lot of tools, devices and handling processes have to be improved to get better process conditions.

3. RECOMMENDATIONS

The experience, learned in device-construction, assembly, cold and hot commissioning, showed us, that some different main objectives are important for a successful work:

- Best quality assurance of the hardware;
- Do all steps in 'cold area' (mock-up) before hot operation;
- Coach your staff in a very sufficient way;
- Look out for interventions and repair opportunities of all equipment;
- Do a very good planning of all operations before in operation and calculate with appropriate tolerances;
- Investigate the material for dismantling as good as possible to prevent from real surprises.

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REACTOR AND FACILITY DECOMMISSIONING EXPERIMENTAL FEEDBACK

(Session 2)

FERMI 1 SODIUM RESIDUE CLEANUP

L. GOODMAN
FERMI 1, DETROIT EDISON COMPANY, UNITED STATES OF AMERICA

Abstract

Fermi 1 was the largest fast breeder reactor in the world when it started up in 1963. The plant was designed for 430 MW•th, though it only operated with a 200 MW•th core. During its operation, people from around the world operated and learned from its experience. Today, the Fermi 1 staff continues to learn during decommissioning. Some of the lessons learned could be valuable to others planning to decommission or design sodium cooled reactors.

1. HISTORY

The three loop reactor was owned by the Power Reactor Development Company. Considerable low power testing was performed during 1963 to 1965. In 1966, during power escalation testing, there was a partial fuel melt event, with no abnormal release to the environment. The cause was determined to be a plate that came loose at the bottom of the reactor, and blocked flow to some fuel assemblies. The plates had been added in an undocumented modification.

Fermi 1 was restarted in 1970 after investigation, cleanup and modifications. The full power test program was completed.

In 1972, the plant was permanently shutdown after consortium members failed to pledge sufficient money to purchase the next core. Detroit Edison took ownership after initial decommissioning since they owned the property and the turbine side of the plant.

2. ORIGINAL DECOMMISSIONING

During the original decommissioning, the fuel and blanket were sent offsite. Primary and secondary sodium were drained. The primary system was left with a purge of CO_2 intended to convert sodium residues to sodium carbonate and ‘passivate’ the sodium. Secondary sodium lines were cut and sealed at the exit of the reactor building.

The secondary sodium in the Intermediate Heat Exchangers (IHX) had been drained to the primary system by drilling a hole in each IHX. The secondary sodium storage tanks and cold traps were sealed after purging with CO_2. Some secondary sodium lines were not well sealed.

The primary cold trap was sent offsite for disposal.

The NaK cooling system was drained, purged with CO_2 and sealed. The primary sodium storage tanks were left with a CO_2 cover gas. Later it was changed to N_2. The fuel transfer tank and reactor overflow tank were purged with air and monitored until there were no indications of reaction. The Table shows how much sodium residues were estimated to remain in the facility. Overall, the sodium systems were considered ‘passivated’. Exactly what ‘passivated’ meant was lost over the years as site management changed.

Fermi 2, a Boiling Water Reactor, was built on the same site. It achieved criticality in 1985. The site focus changed from operating a unique reactor to constructing and then operating a commercial nuclear power plant. Fermi 1 was known to be in SAFSTOR, with its sodium passivated. The terminology led to a general belief that Fermi 1 was in a safe condition, and the sodium was of no concern. Few people were aware that sodium residues were possibly present.

TABLE 1. EXPECTED SODIUM RESIDUES

System/Component	Litres
Lower reactor vessel	284
Transfer rotor container	95
Reactor thermal shielding	4
Reactor rotating plug	189
Offset handling mechanism shield basket	23
Holddown mechanism	15
Primary pumps	30
Reactor blanket plenum	76
15 and 35 cm lines, intermediate heat exchanger (IHX) drain lines	114
76 cm lines	189
Overflow tank	114
IHX seal drain leak-off lines	76
Overflow tank flange shrouds	38
IHX tube bundles	15
Tanks in fuel and repair building	38
Primary storage tanks	189
Service system tunnel piping	189
Secondary sodium system and storage tanks	322
Secondary sodium service system	2025

3. CONDITION OF THE FACILITY

In 1996, Detroit Edison decided to assess the condition of Fermi 1. During explorations in 1997, solid sodium was observed at the bottom of a secondary sodium storage tank. Residues were expected to be sodium carbonate, not sodium.

The as-found condition of the secondary sodium system included three tanks, with ~ 400 L of sodium each, with sodium carbonate at the top tank penetrations. Small bore piping contained various amounts of sodium. Large, poorly sealed pipe had liquid and solid caustic. One cold trap had about 80 L of sodium; the other 2 400 L cold trap was almost full.

The NaK cooling system had some NaK at the bottom of the sump tank, but most of the system was drained. Dams of soda ash were found where the pipe had been cut during original decommissioning. Some liquid NaK was found behind the dams.

The primary sodium storage tanks contained about 200 L of sodium, with some carbonate on interior surfaces.

In the fuel transfer tanks, which had been left open to air in a confined space, there was negligible sodium at the bottom of the tanks. However, sodium was found where pipes penetrated into the tank and in some fuel pots.

The large bore horizontal primary system piping typically has 3 cm or less sodium at the bottom. The large bore vertical pipe is empty. Small bore piping has various amounts of sodium from little to full. Some vent and argon lines had sodium in them. This may have occurred during the draining and siphoning operations during the original decommissioning. The most difficult problem is that the IHX tubes are partly plugged. So far, one of the IHX's has been heated to drain the tubes.

Observations with a camera inside the reactor vessel show most surfaces look relatively clean. Some debris or material was observed on a ledge. The area beneath the core support plates has not yet been observed. In general, more sodium has been found in the plant than expected from the records.

4. PROCESSING

As part of the evaluation of the condition of the facility, the team then evaluated potential cleanup methods, including wet vapor nitrogen (WVN), steam, alcohol, ammonia, and water. The team selected steam in nitrogen as the cleanup process with the best probability of reaching sodium residues and lowest safety risk. It was being used successfully in the non-nuclear industry.

Most of the sodium processing performed at Fermi 1 has been using steam. WVN and water spray have also been used. WVN was used on the NaK cooling system. Water spray was used on the fuel transfer tanks due to the small amount of sodium expected.

So far, approximately twenty tanks or large components and 11 000 kg of small bore sodium containing pipe have been successfully cleaned. A process vessel is used for reacting pieces of small bore piping. Large components are reacted in place. Attached piping is sometimes reacted with the large components, depending on the configuration and if the pipe has a flow path or is blocked. If blocked, steam processing will not be effective. Both water spray and steam were tried unsuccessfully on a portion of the reactor siphon line that passes through the concrete floor. Due to the number of bends in the pipe and the concrete, the pipe cannot be removed. So far, an effective method of clearing the blockage has not been determined.

The basic steam process involves inerting the system, performing a leak test, heating the component in some cases to melt the sodium, and then injecting steam mixed with nitrogen. Hydrogen and NaOH are formed. The gaseous effluent passes through a scrubber and then HEPA filter before being released through a stack. The effluent is monitored for hydrogen and oxygen, and sampled for tritium and radioactive particulates. After the reaction is complete, tanks and other large components are flooded. The NaOH produced from the reactions is neutralized and used to flood other components in lieu of potable water. Currently, sufficient liquid has accumulated, so that other options for the liquid NaOH are being explored. Reuse is preferable.

Setup of the processing systems is time-consuming and expensive. The more complex the system or component being processed, the more complicated the setup is to ensure all surfaces are exposed to steam and sufficient vents are provided.

4.1. Lessons learned

The vent path configuration is important. The vent needs to be large enough that it will not be plugged with NaOH particles. In one case, the sodium vent line did plug with NaOH when there was a right angle elbow approximately one meter above the top of the tank being processed.

A hot reaction or supplemental heat better ensures sodium does not become trapped under the NaOH. Unfortunately, heating also causes hazards. If propane is used, CO_2 is a concern. Hot surfaces can burn people or cause fire retardant materials to ignite, even if electric heat is being used.

Plugs of sodium can remain underneath processed sodium. For example, sodium remained in a secondary cold trap bottom drain line, even though the cold trap was processed with steam and flooded with water.

The primary sodium storage tanks each had a bottom stub. At the end of processing, the procedure required the bottom stub be heated. When heated, the sodium in the stub melted and rose into the NaOH and then reacted, as intended.

When using WVN, droplets condensed in the delivery piping and then dripped into the system. The reaction was between water and NaK, which is stronger and less controlled than a steam or vapor reaction. Also, with WVN, a reaction occurred more than 24 hours after the previous reaction on a couple occasions. It was difficult to determine the end point.

Per the advice of a contractor, the plan was to cut the secondary cold trap into pieces and react the pieces in the process vessel. They had had poor experience in steam processing cold traps in the past, in that sodium remained after processing was completed. Once the mesh of the Fermi 1 cold trap was cut, the sodium in the cold trap started burning to the extent it needed to be extinguished with soda ash. The cutting plan was halted. A cover was welded over the top of the exposed surfaces. The cold trap was successfully processed with steam.

The process vessel used at Fermi 1 was purchased used from a sodium contractor. After approximately 50 batches of pipe, the vessel cracked at the bottom. The cause was determined to be primarily the thermal cycles. During processing, temperatures at the bottom of the vessel reached approximately 600°C. Typically, the vessel would be rinsed after processing with the temperature still above 250°C. Corrective actions were to design and purchase a new vessel with a sacrificial tray that would contain at least most of the sodium being processed. Also, the maximum temperature for the water spray was lowered to reduce the magnitude at the thermal cycles.

Heat needs to be removed from the scrubber, or it will overheat. This is especially true if there is little sodium to react with the steam. Then the steam is mainly carried over to the scrubber. Depending on the amount of heat, the following methods of heat removal have been found effective: feed and bleed or routing the vent through a large tank. During an attempt to process one of the primary loops this summer, overheating of the scrubber occurred such that the feed and bleed operation was almost continuous and could not be reasonably sustained. The process was shutdown. A chiller system is being installed and the gaseous effluents will be routed through the system to allow for condensation. After the physical and procedural changes are made, loop processing will resume.

This issue pointed out the need for sufficient peer reviews in mechanical engineering. The system was reviewed by personnel with chemical engineering and sodium backgrounds, but not mechanical engineering. With a small group, it is most important to insure adequate reviews of planned processes, especially if the group itself does not have broad expertise.

Overall, steam processing experience at Fermi 1 has been favorable. The steam reaction is very definite and can be monitored well. The end point is apparent. The actual reaction phase of processing a batch of sodium containing pipe lasts less than an hour. The actual reaction time for about 200 liters in a large tank was 24 hours.

4.2. Future plans

Remaining sodium processing consists of finishing the first primary loop, reacting sodium residues in the reactor vessel and then the residues in the other two loops.

For the reactor vessel, the equipment on top of the vessel needs to be cleaned off first, so that the graphite can be removed from the rotating plug prior to processing. Equipment removal has started. The graphite may contain condensed sodium vapors, and based on the configuration cannot be effectively processed in place. When the graphite blocks are removed, they will be inspected. If sodium residues are present, the blocks will be processed in the process vessel.

The reactor will be processed with steam, flooded up, and then the liquid will be circulated to try to react any remaining hidden residues. Afterwards, the reactor vessel will be removed. A request for proposal will be issued this fall to determine how the vessel will be removed and who will perform the work. One issue being discussed internally is how clean is clean – or what is the definition of sodium cleanup. The consensus is that the sodium can be considered gone if there would be no additional reaction if the component was flooded with water. The other opinion is that there can be no sodium remaining, even in any crevices. The resolution of this question could impact the approach selected for vessel removal and the cost.

To date, the Fermi 1 decommissioning project has cost (in 2005 US$) approximately $22M, at about $3.5M/year. This will increase. The remaining cost is estimated at $28M. This includes removing the radioactive equipment from site and terminating the license.

One of the aspects causing expenditures higher than originally estimated has been difficulty in access to work areas. Many plant areas containing sodium systems are in confined spaces. These concrete rooms are typically only accessible through a hatchway in the floor, roof, or yard. Considerable money has been spent making additional entrances/exits so that work can be performed safely. Holes have been cut in 1 meter thick concrete walls. Bottoms have been cut out of tanks. A stairway to the reactor building basement was installed through an opening in the floor after removing a storage tank. The additional accesses are needed both for safe exit pathways and to improve efficiency while working in the area. Even with the additional access points, there is a lack of efficiency when working in the confined spaces. The majority of the sodium piping removal and system setup activities are in the confined spaces. Designing better access to equipment areas for future sodium cooled reactors would greatly help both maintenance and decommissioning.

Overall, the sodium cleanup has been more difficult than expected, leading to delays and cost increases. So far the Fermi 1 team is learning as each new system is processed and each new challenge addressed. Continuous learning is one of Detroit Edison’s core values. On the Fermi 1 project, the sodium residue removal is definitely a learning experience.

EXPERIENCE OF PRIMARY COOLING SYSTEM MODIFICATION TO INCREASE HEAT REMOVAL CAPABILITY IN THE EXPERIMENTAL FAST REACTOR JOYO

C. ITO, K. ISOZAKI, T. ASHIDA, K. SUMINO, H. KAWAHARA
JAPAN NUCLEAR CYCLE DEVELOPMENT INSTITUTE, O-ARAI ENGINEERING CENTER, IBARAKI, JAPAN

Abstract

The purpose of the MK-III program is to upgrade the irradiation capability of the JOYO experimental sodium-cooled fast reactor. As a result, the neutron flux density of the core was increased and the reactor thermal power was increased to 140 MWt from the originally designed 100 MWt. To accommodate the increased thermal power, the flow rates of sodium coolant in the primary and secondary systems were increased by 20 and 10%, respectively. Also, all intermediate heat exchangers and dump heat exchangers were replaced with new ones. During this replacement, molten sodium and fuel were retained in the reactor vessel. Consequently, the primary cooling system and cover gas boundaries had to be maintained to prevent impurity ingress to the sodium system. During the replacement, the seal bag method, impurity concentration monitoring of cover gas, and low-pressure control of cover gas were applied to prevent damage to existing components and systems. A roller cutter was used for cutting large diameter pipes to prevent ingress of cuttings. The measures taken to reduce the radiation exposure were a lowering of the surrounding dose rate through the use of temporary shielding, shortening of the operation time near the high dose rate area by first doing thorough training, and the employment of protection equipment to avoid contamination. The replacement of components was completed without major trouble, and methods applied for the replacement proved to be effective in the operation and maintenance of sodium cooled reactors.

1. INTRODUCTION

The experimental fast reactor JOYO at O-arai Engineering Center of the Japan Nuclear Cycle Development Institute has been operated since 1977 as the first liquid metal cooled fast reactor in Japan. From 1983 to 2000, JOYO operated with the MK-II core as an irradiation test bed to develop the fuels and materials for future Japanese fast reactors.

To meet the increasing requirements for the various kinds of irradiation tests, the JOYO upgrading program, designated the MK-III program, was carried out to improve its irradiation capability [1-3]. The main objectives of this program are 1.3 times increase of the neutron flux in the core, increasing irradiation period by shortening the reactor shutdown period, and upgrading of irradiation technology.

The reactor thermal power was increased to 140 MWt from the originally designed 100 MWt of MK-II core. This necessitated increasing the flow rate of sodium coolant in the primary system by 20%. Also, two intermediate heat exchangers (IHXs) and four dump heat exchangers (DHXs) were replaced. The principle of the renovation of the cooling system was to minimize the components replacement, so that renovated components were installed in the place where old components were [4, 5].

2. SYSTEM DESCRIPTION AND DESIGN OF HEAT TRANSPORT SYSTEM UPGRADE

2.1. Design concept of heat transport system upgrade

JOYO has two main heat transport systems (HTSs). The heat generated in the core is transported by primary system sodium, which is circulated by a vertical-type, single-stage mechanical pump. An IHX transfers the heat from the primary to the secondary sodium system, and then the heat is removed to the atmosphere by two DHXs. It was decided to retain the primary and secondary sodium pumps. The increased heat transfer rate was achieved by installing new IHXs and DHXs and by resetting major system operating conditions as follows:

- Increase primary and secondary sodium flow rates by ~ 20 and 10%, respectively, without renovation of pump components other than motors.
- Decrease primary and secondary cold-leg sodium temperatures by 20 and 40°C, respectively. The temperature difference between the reactor vessel (RV) inlet and exit increases from 130 to 150°C, and the difference between the IHX secondary inlet and exit increases from 130 to 170°C.
- There is no change in primary and secondary hot leg temperature because such an increase could significantly affect the structural integrity of components such as the RV and IHX.

The renovated components (see Fig. 1) were selected based on the above concepts. Comparison of the MK-II and MK-III operating conditions at rated power and the major specifications of the renovated components are also shown in Fig. 1.

2.2. Description of intermediate heat exchangers

The IHX, shown in, is made of the type 316FR stainless steel (316FR) and is a vertical shell- and tube-type heat exchanger with a free sodium surface (Fig. 2).

The primary sodium flows into the shell side through inlet nozzles and inlet flow holes and then flows down through the tube bundle, transferring heat to the secondary sodium, which flows up inside the heat transfer tubes. The dimensions of the IHX are ~ 8 m height × 2 m diameter, and the heat transfer area is 363 m^2. The size and heat transfer area are almost the same as the old IHX, even though the rated power increased from 50 to 70 MWt. The design conditions and major specifications are shown in Tables 1 and 2, respectively.

The new IHX is designed to reduce the temperature deviations in heat transfer tubes to reduce corrosion product (CP) adhesion on the primary-side surface of the internal structures and to prevent cover gas entrainment in the sodium. The heat transfer tubes are arranged in triangular form. The distance between the heat transfer tubes and the outer and inner shrouds is slightly smaller than the distance between tubes. This optimizes the primary-side flow distribution and decreases the temperature differences between tubes. Pressure loss on the primary side is decreased by optimization of inlet and exit nozzle shapes. A dipped plate that suppresses the sodium free surface turbulence is installed to prevent cover gas entrainment at increased primary flow rates. Reduction of stagnant flow regions by elimination of thermal shielding plates for the inlet and exit nozzles is aimed at reduction of CP deposition.

The temperature difference between the IHX inlet and exit increased from 130 to 150°C on the primary side and from 130 to 170°C on the secondary side. The newly developed 316FR, which shows improved creep rupture and creep fatigue characteristics at elevated temperature, was used as a major structural material to demonstrate its performance under reactor conditions. Its superior heat resistance is important for coping with the thermal shock increase caused by the increased temperature difference between inlet and exit at plant transient conditions, such as scram, external power supply loss, etc.

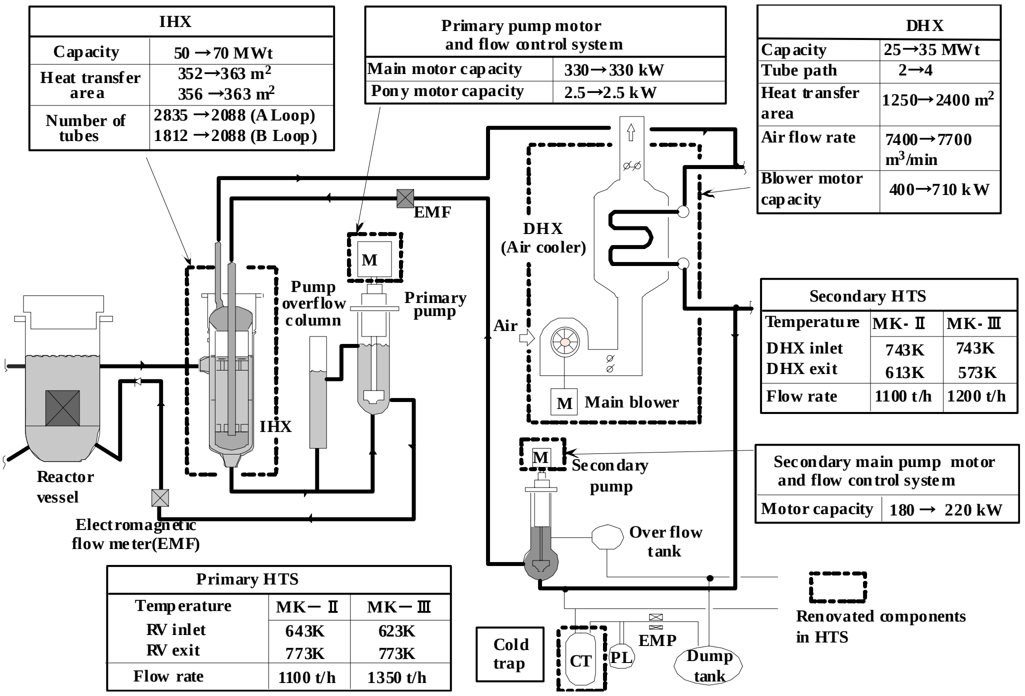

FIG. 1. Schematic of renovated JOYO cooling system and operating condition.

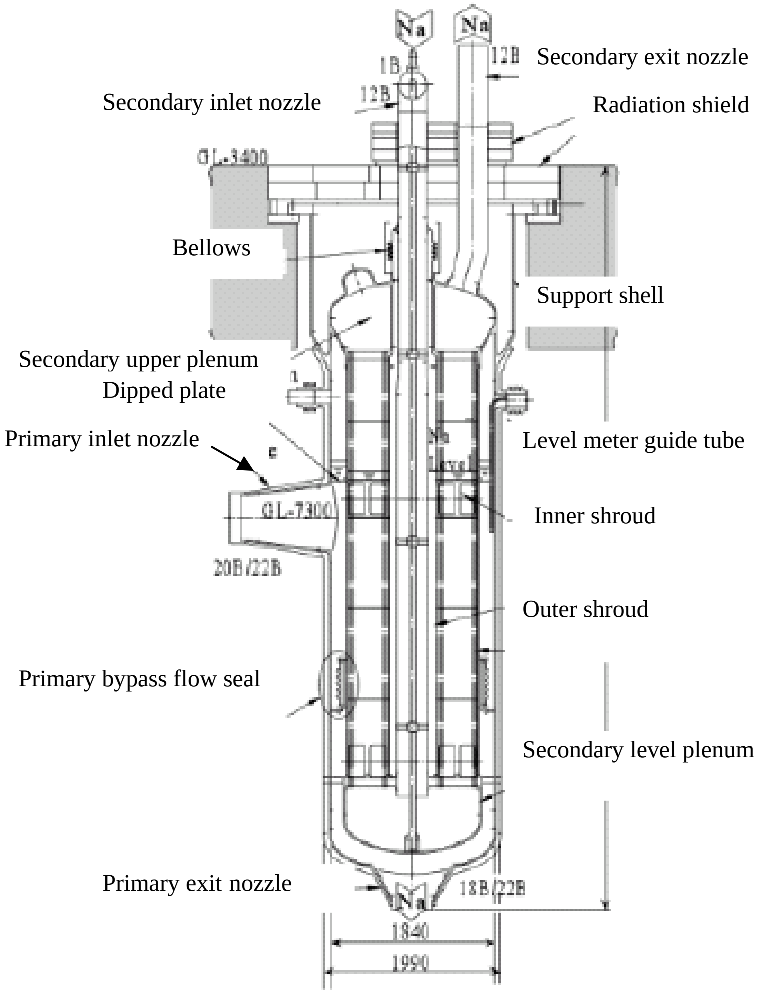

FIG. 2. Schematic structure of IHX for MK-III.

TABLE 1. DESIGN CONDITION OF THE NEW IHX

Type	Vertical-type shell and tube heat exchanger with primary-side free surface	
Rated heat exchange	70 MWt	
Components class	Primary and secondary sides	First-class vessel
	Leak jacket	Third-class vessel
Earthquake-proof class	Primary and secondary sides	As[a]
	Leak jacket	A[b]
Design temperature	Primary	823 K (523 K at negative pressure)
	Secondary	823 K (823 K at negative pressure)
	Leak jacket	823 K
Design pressure	Primary	98 kPa (negative and positive pressure)
	Secondary	490 kPa (positive pressure)
		147 kPa (positive pressure)
	Leak jacket	98 kPa (positive pressure)
		0 (negative pressure)

[a] 'As' is the highest category identified safety in Japanese regulation.
[b] 'A' is the second-highest category identified safety in Japanese regulation.

TABLE 2. MAJOR SPECIFICATIONS OF THE NEW IHX

Property	Value
Heat transfer area (m^2)	363
Unit	2
Dimension	
Primary	
Shell inner diameter/thickness (mm)	1840/19
Secondary	
Plenum diameter upper/lower (mm)	1840/1508
Plenum thickness upper/lower (mm)	19/19
Downcomer diameter upper/lower (mm)	318.5/355.6
Downcomer thickness upper/lower (mm)	10.3/11.1
Tube sheet thickness (mm)	100
Tube diameter/thickness (mm)	19.0/1.0
Leak jacket inner diameter/thickness (mm)	1990/6.0
Radiation shielding diameter (mm)	3260/2830
Radiation shielding thickness (mm)	360/350
Height (IHX-A/IHX-B) (mm)	7870 /8095
Material	
Primary	
Shell and sodium nozzle	316FR, 316FR-F
Bellows	SUS316[a], 316FR
Secondary	
Upper plenum	316FR, 316FR-F
Lower plenum	316FR
Downcomer	316FR-TP
Tube sheet	316FR-F
Tube	316FR-TB
Leak jacket	SUS304, SUS304T[b]
Radiation shielding	SS400[c], SUS304

[a] SUS316 = Type 316 stainless steel.
[b] SUS304 = Type 304 stainless steel.
[c] SS400 = rolled carbon steel for general structure.

3. PLANT CONDITIONS AND REPLACEMENT PROCEDURE

During the upgrade of the HTS, fuel subassemblies were in the RV in order to facilitate decay heat removal, and the RV contained molten sodium with an argon cover gas. For the replacement of the IHXs, sodium in the primary HTS was drained to the primary sodium dump tank while the RV and dump tank were heated to keep the sodium molten. Secondary sodium was drained to the secondary dump tank, and all the electrical heaters except those for the sodium dump tank were shut off in the main HTS during the replacement of the DHXs and pipes. Oxygen ingress to the sodium system had to be prevented because sodium has a high reactivity with oxygen in air, especially when molten. Also, sodium that contains a high concentration of dissolved oxygen corrodes structural material. During the cutting and welding operation for the replacement of components, both primary and secondary coolant boundaries had to remain sealed to prevent air ingress and cover gas ingress.

During the sodium components replacement, several measures were taken to prevent damage to the sodium system. The cutting, installation, and welding sequences were carefully examined and established to minimize opening of the sodium boundary. Furthermore, measures to reduce the radiation exposure and prevent radioactive contamination were applied.

The replacement of components was carried out over about one year with the following plant conditions:

- Reactor vessel containing fuel subassemblies and molten sodium was pre-heated to about 473 K with heated nitrogen gas;
- IHXs and main pipes were kept at room temperature without pre-heating;
- Dump tank that stored sodium was pre-heated with electrical heaters;
- Assisting systems, such as the purification system and sodium over flow system, were kept at room temperature by shut-off of electrical heaters.

During the sodium components replacement, air ingress into the sodium system had to be prevented. The sodium boundary was opened during the sodium pipe cutting, sodium removal in the pipe, and the new and old pipe welding. Replacement sequence and plant operating conditions during the replacement operation were carefully examined and established to prevent air ingress and cover gas egress. The followings procedures were applied to replace the components such as IHX and piping:

(a) Cutting pipe, sodium removal, and old components withdrawal.

  - Withdrawal of the thermal insulator. Temporary withdrawal of obstacles such as pipe, air conditioning duct and conduit;
  - Cutting and withdrawal of IHX connecting primary main outer pipe;
  - Cutting sodium pipe, residual sodium removal, and temporary sealing of sodium pipe opening by seal plug;
  - Withdrawal of components, connecting sodium pipe.

(b) New component installation and welding;

  - New component installation;
  - Half layer welding and inspection of sodium pipe by radiographic test (RT) and liquid penetrant test (PT);
  - Final layer welding and inspection of sodium pipe by RT and PT;
    (i) Welding of IHX connecting primary main outer pipe;
    (ii) Installation of thermal insulator and reinstallation of the temporarily withdrawn obstacles.

Figure 3 shows a picture taken during the installation of a new IHX. Figure 4 shows the cutting section and sequence of IHX connecting pipes. This sequence was chosen based on minimization of the sodium boundary opening, taking into account the installed condition of connecting pipe.

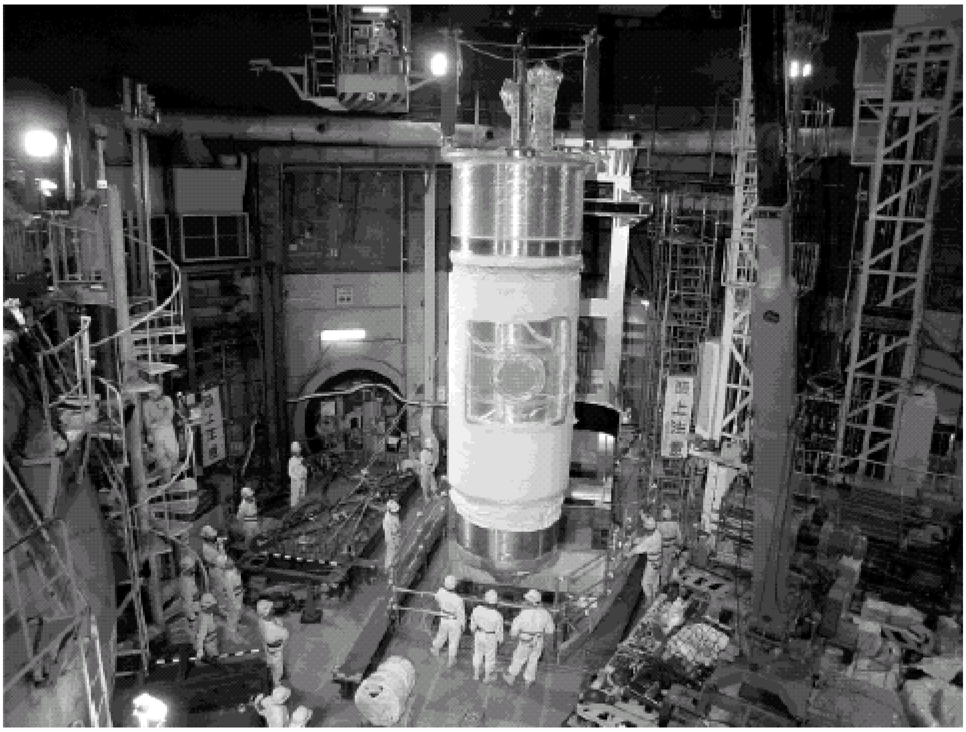

FIG. 3. New IHX installation in the reactor containment vessel.

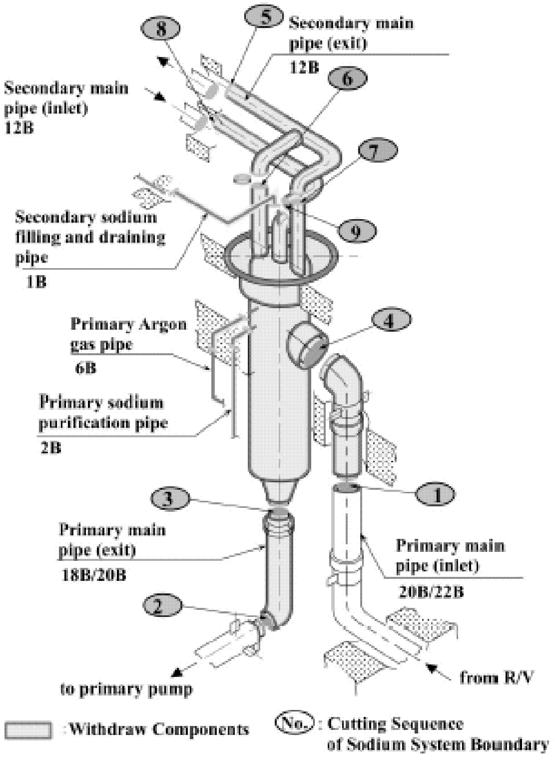

FIG. 4. Cutting section and sequence of IHX connecting pipe (loop A).

4. SODIUM IMPURITY CONTROL

4.1. Prediction of oxygen ingress

Oxygen is the major impurity whose ingress to the sodium system must be prevented. There are three paths of oxygen ingress: oxygen in argon cover gas, oxygen adhered to surfaces of components, and air ingress.

During the replacement, argon gas was supplied to the sodium system to maintain cover gas pressure. The cover gas contained ~ 0.2 ppm oxygen as an impurity, and ~ 2 g of oxygen ingress was estimated. Oxygen adhered to the inner surfaces of the new IHXs made of 316FR and connecting pipes made of stainless steel. Previously measured adhered oxygen on the stainless steel was 0.33 g/m^2, and the inner surface area of new IHXs and primary pipes is 1234 m^2. Thus, the total estimated oxygen that adhered to the new primary components was 408 g. The last path is air ingress to the sodium system during the cutting and welding of pipes.

4.2. Control of oxygen concentration

Operations for which the sodium boundary was opened cutting pipes, sodium removal, and welding pipes were carried out in a seal bag filled with low-pressure argon gas in order to lower oxygen ingress. An image of the seal bag is shown in Fig. 5.

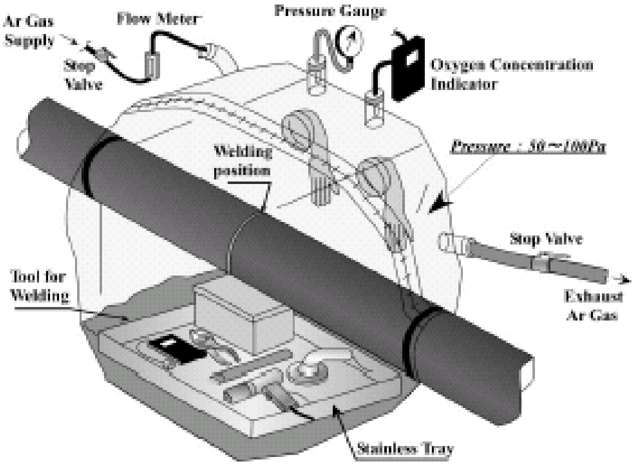

FIG. 5. *Image of seal bag for primary pipe.*

The seal bag was made of a transparent vinyl chloride sheet with a thickness of 0.5 mm. Its design temperature and pressure were 338 K and 690 Pa, respectively. The seal bag was bonded to the pipe by adhesive tape and clamped by hose bands. Also, the seal bag had a hanging hook at the top to prevent slackening. Oxygen concentration in the seal bag was controlled to < 1000 ppm and monitored by a portable oxygen indicator. The control set value for oxygen concentration in the primary system cover was under 300 ppm. These control values were chosen based on previous maintenance experience.

In addition, the nitrogen concentration was monitored and controlled below 1200 ppm. The monitoring considered the concentration ratio of nitrogen and oxygen in air as well as the fact that oxygen is easily reacted with sodium and that it easily dissolves in sodium. Nitrogen, oxygen, and hydrogen concentrations in the cover gas were monitored using gas chromatography. The monitoring data obtained during the welding operation are shown in Fig. 6.

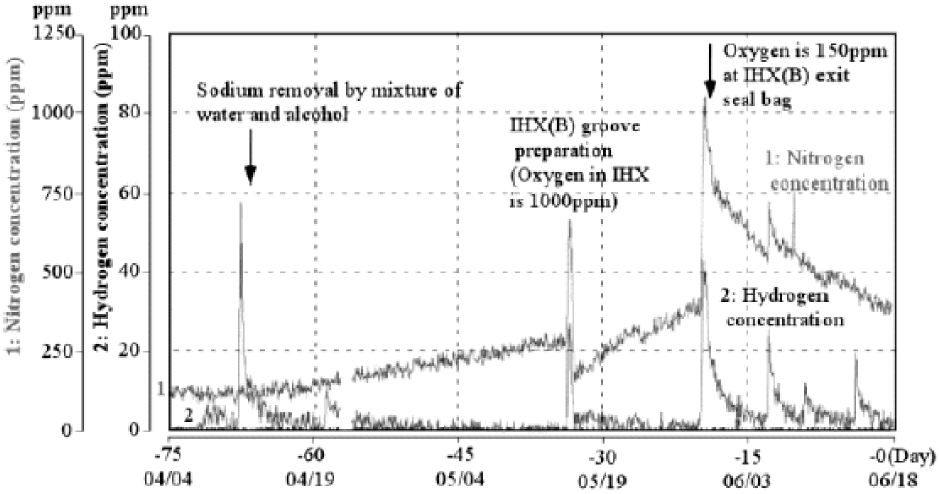

FIG. 6. Nitrogen and hydrogen change during sodium removal and welding.

The hydrogen concentration was not a controlled quantity from the viewpoint of impurity ingress. A hydrogen concentration increase was observed during sodium removal using a mixture of water and alcohol. Hydrogen was generated by sodium-water and sodium-alcohol reactions. During this operation, no nitrogen concentration increase was observed. During other operations, such as groove preparation and welding, both nitrogen and hydrogen concentration increases were observed. In these operations, nitrogen, hydrogen, and water in air could have entered the sodium system from the new IHX and seal bag. Almost no oxygen was observed during the operation, and the maximum nitrogen concentration in cover gas was under 1000 ppm. Based on the oxygen concentration in the seal bag and the operation time with the seal bag, the estimated accumulated oxygen ingress to the sodium system was ~ 180 g, which was one-tenth of the planned value.

5. RADIATION CONTROL

The replacement requested protection from radiation exposure in high-dose-rate surroundings and the very limited accessible space of the radiation controlled area in the containment vessel. Primary sodium contains radioactive 22Na and radioactive CPs, such as 60Co and 54Mn. This sodium adhered to the inner surface of the IHXs and pipes. The highest surface dose rate on the IHX was several tens of mSv0h. The distribution of the surface dose rate is shown in Fig. 7.

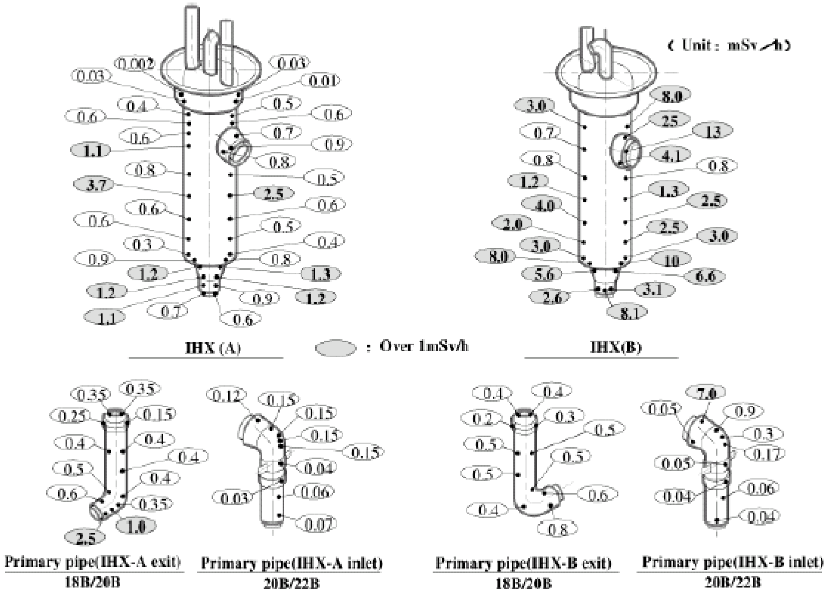

FIG. 7. Surface exposure rate of IHX and connecting pipes.

Furthermore, the distance from the component surfaces for attenuation of the dose rate during the cutting of IHX connecting pipes was less than desired. Finally, to avoid welding flaws, it was necessary to remove residual radioactive sodium that adhered inside the pipe.

There were four measures to reduce the radiation exposure and to prevent radioactive contamination:

- Lowering of the surrounding dose rate;
- Shortening of the operation time near the high-dose-rate surroundings;
- Reducing the surface and internal contamination with protection equipment;
- Employing a seal bag when the primary system was open.

The radiation exposure is limited to an effective dose rate of 50 mSv/year, and the precaution level is 13 mSv/ 3 months or 20 mSv/year. The radiation exposure was controlled based on the principal of as low as reasonably achievable. About 230 lead plates, 170 lead blankets, and 80 lead tamping bags were used as temporary radiation shielding. Lead blankets were placed around the IHX primary inlet and exit pipe elbows as well as the upper and lower IHX shells. Lead plates were placed around the IHX primary exit nozzle and lead plates, and lead tamping bags were placed around the IHX primary inlet nozzle. These places were selected based on high radiation dose rate and proximity to the operation area. After the placement of temporary shielding, the surrounding dose rate in the operation area was reduced ~ 20%.

Before the IHX replacement, mock-up tests using a piping model and a full simulation with a full-sized model were carried out. The objective of the piping model mock-up test was to establish the cutting and welding operation procedure. The full-sized model simulated the IHX primary inlet and exit sections, including the upper and lower parts of the IHX, inlet and exit nozzles, inlet and exit pipes, pedestal, obstacles such as air-conditioning ducts and pipes, and supports. The objectives of this model were to optimize operating methods, to check operability from the cutting to the welding operation with contamination protection equipment, and to measure the operation time for evaluating radiation exposure. Also, this model was used for worker training. The operation time was shortened by 5 to 8% after completion of training.

Several kinds of contamination protection equipment were used to minimize radiation exposure. Rubber gloves and contamination protection suits were used to prevent external contamination. Air-line masks were used to prevent internal contamination as well as to ensure an adequate supply of oxygen. During the primary boundary opening operation, the seal bag served not only to prevent impurity ingress to the sodium system but also to prevent contamination. As a secondary measure to prevent the spread of contamination, a greenhouse made of incombustible transparent sheet equipped with local ventilation was placed around the seal bag.

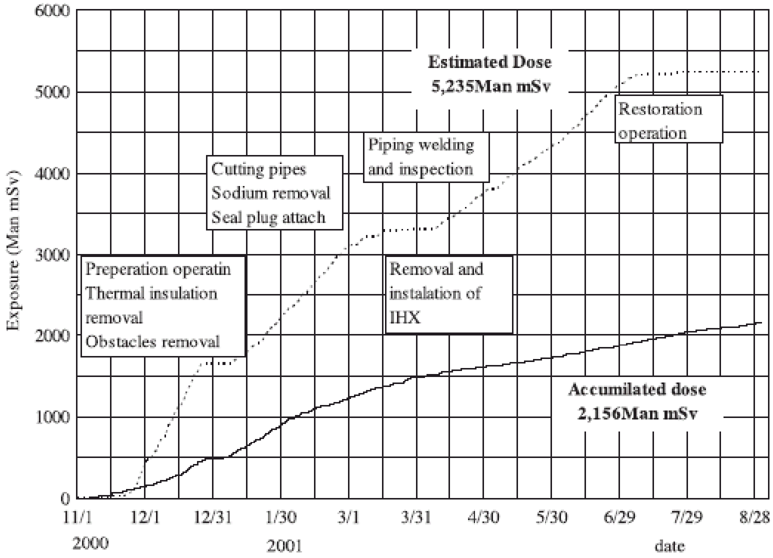

FIG. 8. Radiation exposure during primary system renovation operation.

As a result of these radiation control measures, cumulative radiation exposure was below one-half of the planned exposure. The exposure record shown in Fig. 8 has the planned and actual exposure histories. It should be noted that no one was contaminated.

6. CONTROL OF SODIUM BOUNDARY OPENED

Two kinds of cutting devices were used for the pipe cutting to prevent inflow of cutting pieces. When vertical pipe was cut, the bite was used to cut a two-thirds layer in atmosphere then the residual one-third layer was pressed down by the roller cutter in the seal bag under argon gas. On the other hand, horizontal pipe was cut off by the bite and cutting pieces in the pipe were removed together with sodium adhered on the inner surface. A seal bag was placed after cutting a two-thirds layer. During the sodium removal in the pipe, a stopper was inserted into the vertical pipe to prevent dropping of tools. In addition the number and cleanliness of tools were correctly checked before and after each operation.

During the MK-III reactor operation, primary and secondary cover gas pressure is controlled to about 500 Pa and 40 kPa, respectively. Argon cover gas pressure was controlled to levels lower than normal reactor operation during the renovation operation. Between cutting and welding pipes operation, the sodium boundary tightness was kept by the temporary closing equipment seal plug. During the replacement operation, a seal bag made of transparent vinyl chloride sheet and glove box were used to prevent air ingress. The pressure resistance of the seal bag was 600 Pa. Argon cover gas pressure was controlled to below 300 Pa, considering operability in the seal bag and to prevent seal bag damage.

7. WELDING OPERATION

7.1. Sodium removal

Adhered sodium on the inner surfaces of used pipe in the sodium system was removed to avoid welding flaws between used and new pipes. Figure 9 shows measures to remove sodium.

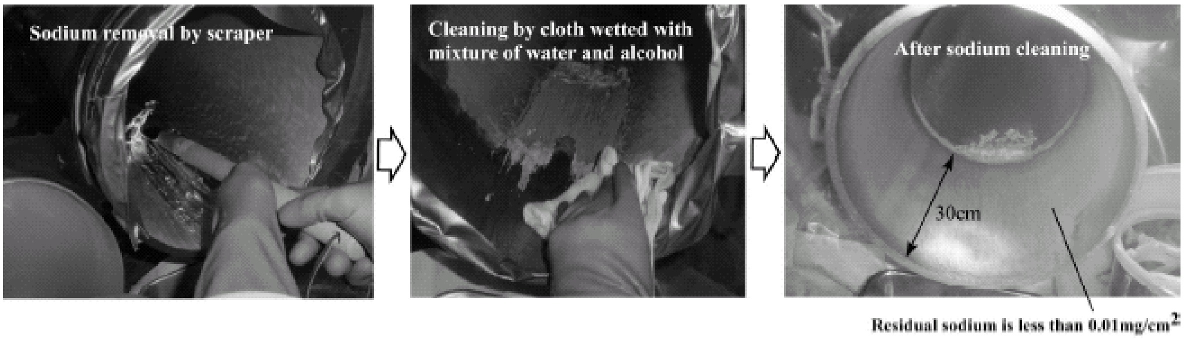

FIG. 9. Radiation exposure during primary system renovation operation.

When the pipe, especially a small diameter pipe, was choked with sodium, a spatula was used to scrape off the sodium. Large deposits were removed using a scraper and drill, while small amounts of sodium were removed using a cloth wetted with a mixture of water and alcohol that contained 50 to 70% ethyl alcohol. Residual sodium after cleaning was less than 0.01 mg/cm^2.

To avoid sodium melting by the welding heat, the length of sodium removal in the pipe was about 30 cm from the opening and the surface temperature of used pipe was controlled to less than 343 K during welding.

7.2. Control of back seal gas

Tungsten inert gas (TIG) welding was used for the pipe welding. During the TIG welding, back seal gas flow is necessary to avoid welding flaws caused by oxygen, nitrogen and hydrogen ingress to the metal melting zone. The argon cover gas was only used as back seal gas. Mock up tests to verify the welding condition showed that the pressure difference between the cover gas and the gas inside seal bag had to be kept under 100 Pa. Based on the result of mock up tests and operability of seal bag, both cover gas and seal bag gas pressure was controlled to between 50 and 100 Pa during the welding operation.

8. RESULTS

8.1. Functional test results

Upon completion of the replacement, integrated functional tests were carried out to verify the success of the replacement operation and the basic function of replaced components at low sodium temperature without reactor operation. The major functional tests for the primary system were sodium impurity measurement, characterization test of the primary pump and vibration test of the primary loop.

Removed oxygen was 383 g during purification operations in the integrated functional test. This is ~ 60% of the predicted value, which was estimated in advance based on the three paths of oxygen ingress. Therefore, it was confirmed that impurity ingress to the primary system was successfully controlled during the replacement period.

The functional tests, which addressed issues beyond heat transfer performance, showed that the design of components and replacement work were satisfactory.

8.2. Lessons learned

Large sodium components were replaced under difficult conditions: limited work space during operations such as cutting and welding in the existing nuclear plant; primary sodium coolant boundary opening with fuel subassemblies and molten sodium in the RV; high radiation dose rate and treatment of radioactive sodium.

Replacement operations were successfully completed despite these difficulties. The following lessons learned through these operations will be useful for future maintenance and modification of sodium-cooled fast reactors:

- Application of seal bags during sodium pipe cutting, sodium removal, and welding is useful to prevent oxygen ingress to the sodium system, to avoid a deficiency of oxygen in the working area, and to minimize contamination during boundary opening.
- Cover gas and seal bag gas pressure control below 100 Pa is necessary to prevent welding flaws and is useful during opening of a sodium boundary to have enough pressure margin for the seal bag and good operability.
- On-line measurement of nitrogen and hydrogen in the cover gas system is effective to monitor impurity ingress to the sodium system.
- Sodium adhered on the inner surfaces of pipe was effectively and safely removed by a mechanical scraper or drill and a cloth moistened by a mixture of alcohol and water.
- Mock-up tests are useful both to shorten the operation time in high-dose-rate surroundings by verifying the applicability of operating procedures and to train workers.

9. CONCLUSIONS

After successful replacement and functional testing, JOYO has been put into operation for performance tests. Lessons learned through the replacement operations will be fully applied not only to future JOYO maintenance and modification activities but also to future sodium-cooled fast reactor design, construction, and operation.

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MACHINING EXPERIENCES OF THE PRIMARY VESSEL WITH RESIDUAL AMOUNT OF SODIUM

J. RUSS
WESTINGHOUSE ELECTRIC GERMANY GMBH, MANNHEIM, GERMANY

Abstract

Westinghouse was assigned with the decommissioning of the KNK II plant by the Research Centre Karlsruhe. The development of the decommissioning methodology was performed in co-operation with the KNK II and different industrial organizations. The most substantial subject, as the decommissioning of the reactor vessel, is performed under specific boundary conditions as residual sodium in the vessel on nitrogen environment.

1. INTRODUCTION

The goal of this presentation is to focus on the obtained experiences of the decommissioning of the primary vessel at KNK-II with regard to the design and tested process and equipment. The interconnection between the real conditions on site and the qualified process and equipment and its influences on the resulting changes to the process and equipment are also described.

2. DESIGN AND EXPERIENCES

2.1. Design boundary conditions

The major design parameters for developing the equipment, tools, process and the tests have been established under the given boundary conditions as follow:

- Material embrittlement due to neutrons (E>1 MeV of 2.8E+20 1/cm²);
- Equipment vibrations varying with the cutting parameters;
- Sodium on the vessel parts to be machined;
- Sodium inside the pipe nozzles to be cut from the vessel;
- Equipment design under radiation consideration

2.2. Experiences

2.2.1. Material and vibrations

The test material, simulating the embrittled vessel material, was selected based on the material life time samples of the vessel.

- For equipment, process design and tests a material with an ultimate strength of 560 N/mm² was selected;
- In reality the embrittled material for the Reflector has an estimated ultimate strength about 700 N/mm² and the Thermal Shield 600 N/mm². This value is estimated based on the recorded machining torque.

The actual Reflector material is about 30% more embrittled than the simulated one for tests. This leads to an increase of the milling torque by the factor 2. This divergence substantially influences the machining process in terms of machining parameters and vibration. The milling stroke speed had to be reduced in an area of 30 to 70% of the value established during the test. The milling speed at this high load level is depending on the milling direction influenced by the stiffness of the machine. Due to different resonance frequencies of the machine elements and the vessel pieces, it was required to change the established milling rotation speed adequately. The changes were performed in an area of +/-40% of the nominal value. Both parameters have to be adapted by the operator based on the visual and optical evaluation of the vibration behaviour. Due to the requested large range of the rotation speed, the motor of the driving milling module had to be replaced by one with a higher rotation speed. At this point in time experiences are collected with these improvements.

Another parameter which had to be adapted was the axial clamping force of the millers. The design value of 8 KN was strong enough for the material with an ultimate strength of 560 KN. For the real embrittled material in combination with the real vibration behaviour, the clamping force had to be increased up to 10 KN. The Nozzles, the Thermo Shock Liner as well as the Thermal Shield were simulated very close to the reality in the test, so that the established milling parameters could be applied on scale 1:1. With the obtained experiences, the vibration behaviour could be controlled by adapting manually the milling parameters.

2.2.2. Residual sodium and contamination

As mentioned in chapter 2.1, two cases of residual sodium were considered: Sodium on the surfaces of the vessel and sodium inside the nozzles. For safety reasons due to atmospheric test condition and milling temperature above the melting point of the sodium, residual sodium on the surfaces was not implemented in performed tests. On site, it has been experienced that the influences of the residual sodium on the mechanical machining process can not be neglected. The developed machining process is mainly performed with finger millers for thicker walls and disc millers for thin wall pieces. The disk miller process is less sensitive to residual sodium, while the sodium is influencing the finger milling process drastically. During the milling process the temperature is rising higher than 90°C, the sodium melts, the melted sodium is getting mixed with the milling chips and after cooling down, this mass consisting in sodium sticky chips is blocking the chip channels on the miller. Due to this phenomenon the milling heat can not be transferred anymore away from the miller and the temperature on the miller increases up to its melting point. With the obtained experiences, the described phenomenon could be eliminated by adapting the dismantling strategy and controlling the milling torque.

The melted residual sodium during milling on the inner surface of a cut piece has less influence than the residual sodium on the outer surface. The melted sodium on the outer surface is transferred with the chips inside the milling gap where the sodium and the chips cool down and the chips are blocked as described above. The worst location of residual sodium is a gap which is crossing the milling gap. The amount of sodium is much higher in such gaps than on surfaces and adequately the risk for blocked chips, too. A developed nitrogen cooling process of the millers had to be abandoned due to radiation consideration. With the developed cooling process, flying chips were produced on the machine. These chips on the machine can lead to a drastic increase of the radiation of the maintenance personal.

As mentioned above the second case of residual sodium is inside the nozzles. To get access with the disk cutting tool inside the nozzles, the residual sodium has been removed by a finger miller with 3 large chip grooves. The tested process and tools could be applied scale 1:1 on side.

3. SUMMARY AND RECOMMENDATIONS

In summary it can be concluded that major factors which are affecting the dismantling process of the primary vessel at KNK-II is the underestimated embrittlement of the material especially that of the reflector, and the residual amount of sodium on the gaps and on the outer surfaces of the dismantled pieces. These two factors were the main driving forces for improvements on site. By implementing these improvements the decommissioning of the KNK-II primary vessel is continuing successfully. With the obtained experiences up to now, for a further project it’s recommended to implement:

— Equipment with a remote controlled anti-vibration system;
— Process and equipment design for worst case embrittled material;
— Tool and process design, which is less sensitive against residual sodium;
— An as-built online recording system.

I would like to express many thanks to all parties implemented in the improvement of the process. Only with their support the decommissioning work is proceeding successfully.

DECOMMISSIONING OF EXPERIMENTAL BREEDER REACTOR–II COMPLEX, POST SODIUM DRAINING

J.A. MICHELBACHER, S.P. HENSLEE, C.J. KNIGHT, S.R. SHERMAN
IDAHO NATIONAL LABORATORY, IDAHO FALLS, USA

Abstract

The Experimental Breeder Reactor – II (EBR-II) was shut down in September 1994 as mandated by the United States Department of Energy. This sodium cooled reactor had been in service since 1964. The bulk sodium was drained from the primary and secondary systems and processed. Residual sodium remaining in the systems after draining was converted into sodium bicarbonate using humid carbon dioxide. This technique was tested at Argonne National Laboratory in Illinois under controlled conditions, and then demonstrated on a larger scale by treating residual sodium within the EBR-II secondary cooling system, followed by the primary tank. This process, terminated in 2002, was used to place a layer of sodium bicarbonate over all exposed surfaces of sodium. Treatment of the remaining EBR-II sodium is governed by the Resource Conservation and Recovery Act (RCRA). The Idaho Department of Environmental Quality issued a RCRA Operating Permit in 2002, mandating that all hazardous materials be removed from EBR-II within a 10 year period, with the ability to extend the permit and treatment period for another 10 years. A preliminary plan has been formulated to remove the remaining sodium and NaK from the primary and secondary systems using moist carbon dioxide, steam and nitrogen, and water flush. The moist carbon dioxide treatment was resumed in May 2004. As of August 2005, approximately 60% of the residual sodium within the EBR-II primary tank had been treated. This process continued through the end of 2005, when it became increasingly ineffective. At that time, subsequent treatment processes were initiated. It should be noted that the processes and anticipated costs associated with these processes are preliminary. Detailed engineering has not been performed, and approval for these methods has not been obtained from the regulator or the sponsors.

1. INTRODUCTION

During the EBR-II Plant Closure Project, initial deactivation of residual sodium was performed using humidified carbon dioxide. A different terminology was used at that time, and the deactivation process was referred to as ‘residual sodium passivation.’ Since the deactivation process creates a solid carbonate layer on top of the residual sodium, an analogy was made between this ‘oxide’ layer, and the formation of oxide layers on other metals, such as aluminium. In the case of aluminium, for example, the oxide layer that forms on the surface passivates it, and prevents any further reaction of oxygen with the metal underneath. In the case of sodium, the carbonate layer does not prevent any further reaction of humidified carbon dioxide with the residual sodium metal underneath, and so technically does not passivate the residual sodium. It does, however, provide a resistance barrier to gaseous diffusion and would slow the reaction of gaseous reactants with the sodium underneath. Such a resistance barrier would be useful in the case that uncontrolled leakage of moisture occurs into any of the EBR-II sodium systems during the potentially long period of time between the finish of the Plant Closure Project and the resumption of residual sodium deactivation. Placing a thin layer of carbonate on the residual sodium surfaces in the EBR-II sodium systems, the Project was able demonstrate partial deactivation that was acceptable to the Department of Energy in regard to the stored residual sodium, so that the EBR-II sodium systems could be placed in a steady state condition indefinitely while awaiting further resources (i.e., funding, personnel) to resume full sodium deactivation.

2. BACKGROUND

The EBR-II was a sodium cooled research reactor located in the south-eastern portion of the Idaho National Laboratory. The EBR-II was a 62.5 MW thermal reactor that began operations in July 1964, and when fully operational, produced 19.5 MW of electrical power for the INL electrical grid. The EBR-II complex is presented in Fig. 1.

FIG. 1. EBR-II Complex.

It consists of the reactor and reactor building, the Sodium Boiler Building, the electrical power plant, reactor cooling towers, water chemistry laboratory support facilities, and the cover gas cleanup system. The EBR-II reactor building is connected, through a below grade tunnel, to the Fuel Conditioning Facility, a large inert atmosphere hot cell facility. The Fuel Conditioning Facility hot cell is used to support sodium bonded fuel treatment and research. The reactor building, a cylindrical structure with a hemispherical domed top, has a steel containment shell with an inner diameter of 24.4 m (80 feet) and a height of 42.4 m (139 feet). The bottom and sides are 2.5 cm (1 inch) thick steel plate and the dome is 1.3 cm (2 inch) thick, lined with a 10.2 cm (4 inch) concrete missile shield.

The reactor was a test facility for fuels development, materials irradiation, system and control theory tests, and hardware development. The EBR-II core and blanket subassemblies were contained within the reactor vessel (Fig. 2) prior to defueling.

The 1.70 m (67 inch) diameter vessel and its shield were immersed in a sodium pool within the 7.9 m (26 feet) diameter by 7.9 m (26 feet) high primary tank. The sodium contained within this tank represented the primary cooling system for removal of the heat from the reactor core. The primary system contained about 325 m^3 (86 000 gallons) of sodium, and transferred heat to the secondary sodium system (which contained about 50 m^3 (13 000 gallons) of sodium) through a sodium-to-sodium intermediate heat exchanger that was immersed in the primary sodium. The secondary sodium was circulated in a closed loop through superheaters and steam generators outside of the reactor containment in the Sodium Boiler Building. The high pressure steam produced in the steam generators drove a turbine-generator to produce electric power.

The EBR-II Plant Closure Project’s charter was to maintain the facility and complex in an industrially and radiologically safe condition. This was accomplished via the closure plan, contained in the Environmental Assessment [1] for the Shutdown of Experimental Breeder Reactor – II at Argonne National Laboratory – West. This Environmental Assessment was granted a Finding of No Significant Impact by the United States Department of Energy, documenting that the proposed actions would not constitute a major federal action significantly affecting the quality of the human environment, negating the necessity for the preparation of an Environmental Impact Statement.

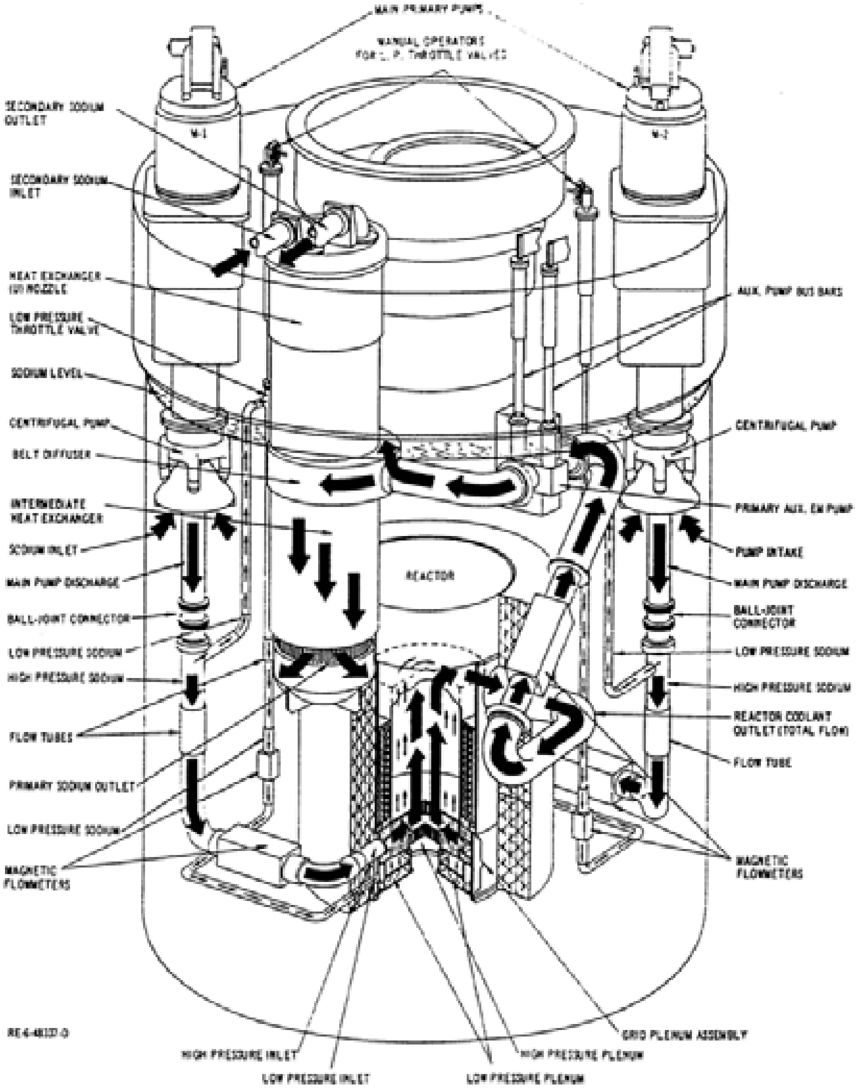

FIG. 2. EBR-II Primary tank.

Closure of the EBR-II complex was achieved through completion of reactor defueling, primary tank draining, processing of the primary, secondary, and Fermi-1 primary sodium, residual sodium passivation, and system lay-up. Reactor defueling consisted of removal of all the fuelled assemblies in the 637 core positions and replacing them with non-fuelled assemblies of the same configuration. The fuelled assemblies were packaged for storage and subsequent treatment.

The primary tank was not originally designed to be drained. A draining system was designed and fabricated using an annular linear induction pump, and operated to drain the primary tank to the secondary storage tank in ~ 40 m^3 (10 000 gallons) batches. This sodium was then transferred to the Sodium Process Facility through a pipeline and reacted to 70+ wt % sodium hydroxide for disposal. The Sodium Process Facility was designed, constructed, and operated for the processing of the EBR-II primary and secondary sodium, as well as the Fermi 1 primary sodium that was stored at the INL.

3. RESIDUAL SODIUM

3.1. Test program

Laboratory experiments were performed at Argonne National Laboratory in Illinois to study the effects of humidified carbon dioxide on samples of sodium metal. The goals of the experiments were to determine the stability of the deactivation process in regard to sudden temperature and pressure changes and changes in the hydrogen generation rate, confirm the chemical composition of the sodium bicarbonate layer, observe the morphology and measure the density of the sodium bicarbonate layer, and determine the deactivation rate as a function of sodium bicarbonate layer thickness and moisture input rates.

In these experiments, samples of sodium metal were placed into a test chamber and exposed to a continuous flow of humidified carbon dioxide. The concentrations of hydrogen and oxygen in the exhaust gas were measured and recorded. The growth of the sodium bicarbonate layer on the samples was observed over time and measurements were taken on the growth rate of the layer versus the consumption rates of the sodium samples. Observations were recorded concerning the morphology of the sodium bicarbonate layer and samples were taken of the layer for chemical analysis. Details of the experimental equipment, procedures, and experimental results and analyses are presented in Ref. [2].

These experiments demonstrated that the deactivation of residual sodium metal can be performed safely and in a controlled manner with humidified carbon dioxide. The use of humidified carbon dioxide to deactivate residual sodium results in the transformation of sodium metal into a solid, powdery, white colored material that is primarily composed of sodium bicarbonate. A decreasing sodium reaction rate can be correlated with an increasing thickness of the sodium bicarbonate layer. Expansion of the sodium bicarbonate layer can result in the breakage of glassware and the deformation of sheet metal in confined spaces as was demonstrated under test conditions. For the sample with the longest exposure, a reaction depth of 5.5 cm of sodium was achieved, with demonstrated penetration of water vapor through 21 cm of sodium bicarbonate.

3.2. Secondary system

The deactivation of the EBR-II secondary sodium system residual sodium was used to qualify the humidified carbon dioxide deactivation technique on a full-scale system, and to gather safety and performance information so that the technique could be applied to the deactivation of residual sodium within the EBR-II primary cooling system.

The bulk quantities of the secondary sodium were drained into the secondary sodium storage tank and pumped to the Sodium Process Facility for disposition. The secondary sodium system was then modified from its original configuration in order to perform the initial sodium deactivation. First, a carbon dioxide header was installed to channel the flow of carbon dioxide in seven different directions. Modifications were then made to the secondary sodium cooling system to allow for the creation of 14 different gaseous flow paths. The flow paths were not isolated, however, and there was a certain degree of overlap in the flow paths due to the highly interconnected nature of the secondary sodium cooling system components. Also, a vent line was installed at a central location so that all of the exhaust lines could be purged. A hydrogen monitor, an oxygen monitor, and a sample gas-conditioning unit, all identical to the ones used for the laboratory experiments, were installed on the vent line.

The deactivation of the residual sodium in the secondary cooling system was performed in two phases. During the first phase, each of the 14 flow paths were exposed to humid carbon dioxide for a relatively short period of time (from two days to two weeks) in order to react the most accessible residual sodium. During the second phase, a more concentrated deactivation of the residual sodium within a specific superheater pathway was performed because the superheater was known to contain a deeper pool of residual sodium, and the Project wanted to perform a visual examination of the deep pool after treatment. During the first phase, approximately 115 L of water were evaporated from the carbon dioxide humidification cart. In this cart, dry carbon dioxide was bubbled through a height of approximately 1 m of water to humidify the carbon dioxide. This 115 L of water is sufficient to react up to 145 kg of sodium. Since only 92 kg were reacted according to an integration of the hydrogen concentration data, some water must have been lost from the system through the vent manifold. This water loss was confirmed by the observation that water was frequently collected from the sample gas conditioner unit.

In the second phase, deactivation of the superheater pathway was resumed and treated for an additional 72 days. According to integration of the measured hydrogen data, another 90 kg of sodium metal was consumed. Approximately 110 L of water were evaporated from the carbon dioxide humidification cart during the conduct of the second phase, which is consistent with the amount of water evaporated from the cart during the first phase for a similar period of time and amount of sodium reacted.

No end point was reached in the treatment process, and residual sodium still remains in the superheater. Treatment was stopped on the superheater because sufficient information had been collected to certify that the treatment process was safe and to verify that the equipment and instrumentation were reliable enough to monitor the treatment process. Further treatment will be needed to fully react all of the residual sodium within the superheater.

Following the shutdown of residual sodium treatment operations in the superheater, three holes were drilled in the side of the superheater near the level of unreacted sodium in the bottom of the superheater. The first hole was drilled 2.5 cm above the sodium level and revealed only white power. The second hole was drilled 2.5 cm below the sodium level and revealed only solid sodium metal. The final hole was drilled at the sodium level and revealed a solid mixture of white powder above sodium metal. These observations confirmed that the carbonate material accumulates on the exposed surface of sodium metal and does not significantly penetrate the sodium layer beneath the surface.

3.3. Current state and future treatment options

Since completion of the initial treatment discussed above, the secondary sodium system has been maintained under a dry carbon dioxide blanket maintained at positive pressure. Carbon dioxide is supplied only to counter the leak rate of carbon dioxide from the system. The system is monitored for hydrogen and oxygen, with none detected to date.

Different treatment processes are being recommended for removing the remaining sodium from the secondary system and hence obtaining RCRA closure. The sodium will either be drained, reacted in place, or the component or section containing sodium will be removed from the system and treated elsewhere.

Significant pools of sodium remain in the bottom of the superheaters and evaporators. Drain connections will be welded to these components, the components heated, and the sodium drained to suitable containers. Pools contained in large diameter piping may also be managed using this method.

Narrow pipe sections, dead end pipe legs, and other such parts of the system may require complete removal from the system to reach the residual sodium due to the restrictions to the flow of moist gas. For parts removed, the sections will be cut into manageable pieces and reacted by exposing to a liquid water spray in the INL’s Sodium Components Maintenance Shop (SCMS). This facility is equipped with a large reaction chamber that is capable of treating up to 2.3 kg of sodium at a time, in any configuration. The by-product of this treatment process is a caustic solution, which is recycled and used continuously until the concentration level reaches between 5 and 15wt% hydroxide, at which time it is removed from the system, reacted to carbonate, and solidified in drums for disposal.

The treatment in place will involve further use of humidified carbon dioxide and possibly humidified nitrogen and/or steam-and-nitrogen. This will be decided on a case-by-case basis depending on the configuration of the system. The choice of technique will also depend on factors such as cost and the availability of funding, safety, isolation capabilities, rate of treatment desired, etc.

Following deactivation, all remaining sections will be flushed with liquid water. This flushing operation will be repeated as necessary to react all remaining residual sodium and dissolve all sodium reaction chemical by-products such as sodium hydroxide and sodium bicarbonate. It is expected that this flushing operation will generate up to 60 000 L of solution for further treatment by methods such as neutralization, evaporation, encapsulation, etc. Instead of directly treating and disposing, the waste solution may be saved and used to flush the EBR-II primary system.

3.4. Primary system

Bulk primary sodium has been drained and treated in SPF. Approximately 1.1 m^3 (300 gallons) of liquid sodium remained in the primary tank after draining. In early 2002, moist carbon dioxide was introduced into the tank to produce a sodium carbonate/bicarbonate layer (approximately 2 mm thick). Since completion of this initial carbonation process, a dry carbon dioxide blanket was maintained in the tank until the process was resumed in May 2004.

3.5. Proposed treatment/disposal

The residual sodium remaining in the tank is currently being treated with wet carbon dioxide and, as of August 2005, approximately 60% of the residual sodium has been treated.

Some of the systems/components associated with the primary system contain captivated volumes of sodium isolated from the primary tank. These include the immersion heaters, low pressure plenum throttle valves, primary sodium transfer system, and shutdown cooler bayonets, and will be removed and treated outside the reactor building in SCMS. Some reactor components, including the primary tank-auxiliary heater system, will initially be left in service to assist in residual sodium deactivation.

After treating as much of the residual sodium as possible with moist carbon dioxide, it is anticipated that an intermediate process, using steam and nitrogen, would follow. The primary tank would then be flushed with water to verify that all residual sodium has been deactivated. During flushing of the primary tank, a combination of spraying or heating the water may be used to ensure adequate coverage of the tank and related hardware.

To prepare the primary tank for a water flush, equipment must be installed to provide for introduction and the likely recirculation of the flush water. It is anticipated that this treatment system will also include a means for neutralization of the flush fluid. Sampling and control systems must also be included. In addition, it will probably be necessary to install some form of level indication in the tank to control filling, and limiting the rate of water influx when the water level approaches known pockets of sodium.

Current RCRA regulations require the treatment residuals to be removed from the system. To accommodate this requirement, equipment must be installed in the recirculation system to separate the treatment residuals from the flush water. After the completion of rinsing and meeting RCRA closure performance standards, the treatment water will be removed from the system. It is proposed that the water will be pumped out of the tank and treated in an evaporation/filtration system yet to be designed and constructed. After the treatment water has been removed, the primary tank will be heated to evaporate residual water left in the tank.

4. MISCELLANEOUS COMPONENTS/SYSTEMS

Reference 3 contains a comprehensive list of the equipment and systems that will require additional action in order to achieve RCRA closure. Included are the primary tank cover gas system, NaK transmitters, secondary sodium purification system, intermediate heat exchanger, rotating plugs, primary tank heaters, primary tank nozzles, fuel handling systems, primary tank cover gas sampling supply system, shutdown coolers, reactor building storage pit and manipulators, cover gas cleanup system, primary purification systems, radioactive sodium chemistry loop, reactor building storage holes, argon purge system, argon cooling system, hydrocarbon analyzer, failed fuel transfer system, fuel element rupture detection, and secondary sodium recirculation system. Preliminary plans for removal, as necessary, and sodium reaction have been formulated as the basis for the cost estimate presented below. Additional engineering will be required in order to establish the basis for funding requests from the Department of Energy.

5. PRELIMINARY COST ESTIMATE

A preliminary cost estimate was compiled for achieving RCRA closure on the EBR-II facility. This estimate assumed that the work would be performed within eight years. Assumptions were made without performing detailed engineering analyses or obtaining approval for the proposed actions, leading to uncertainties in the estimates.

Twenty three primary tasks were identified when compiling the cost estimate. For each task, costs were divided into three different categories – materials & supplies (M&S), exempt effort, and non-exempt effort (for the purpose of this paper, the latter two have been combined). Also, the tasks were divided into three separate phases – planning, execution, and treatment/disposal. In addition, a base level of staffing is described that provides the necessary support for all project tasks over the lifespan of the project. The Argonne National Laboratory Pricing Guide was used either directly or indirectly to calculate M&S, effort, and contingency costs that are fully burdened with all applicable divisional and laboratory taxes. Since the project will be completed under Idaho National Laboratory policies and procedures, a different set of pricing guidelines will be implemented for the final estimate. All costs are presented in FY 2004 dollars. Prior to initiating the efforts, a detailed engineering study will be performed, complete with accurate cost estimates associated with each activity. Once internal approvals are obtained, this study will be submitted to the funding agencies to request budgetary approval for proceeding with the closure activities and the project will be staffed accordingly. The following cost estimate is based on a series of assumptions and is presented for information only:

Materials and services	$3 000 000
Labor	$25 000 000
Staffing	$7 000 000
Total w/o contingency	$35 000 000

6. CONCLUSIONS

The EBR-II reactor and systems are required by law to be removed from the RCRA registers by 2012, with the option to extend the completion by up to 10 additional years. A preliminary plan has been formulated to complete the work required by RCRA in eight years. Additional effort is necessary to finalize this plan and obtain funding for completion of the RCRA required tasks.

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SFR WASTE CHARACTERIZATION AND TREATMENT

(Session 3)

DECOMMISSIONING OF CREYS-MALVILLE NUCLEAR POWER PLANT: RADIOLOGICAL CHARACTERIZATION

S. TONTINI
AREVA /FRAMATOME-ANP, LYON, FRANCE

B.R. PONCET
ELECTRICITE DE FRANCE (EDF) – CIDEN, VILLEURBANNE, FRANCE

P. TRABUC
COMMISSARIAT A L'ENERGIE ATOMIQUE (CEA), CADARACHE,
SAINT PAUL LEZ DURANCE,
FRANCE

Abstract

Creys-Malville French fast reactor was shut down a few years ago and it has now to be dismantled with the first EDF NPP. An essential stage in the decommissioning of the Creys-Malville NPP, and of any other nuclear decommissioning project, is the radiological characterization of the installation. Good knowledge of the initial radiological state of the reactor block is essential in order to ensure that the radiological risk is managed during the dismantling operations, and to provide the data necessary for subsequent waste management. Therefore, radiological characterization has to provide the best possible information on: Total and the specific activity of the radio-elements present in the installation and the evolution of this activity in relation to the decay time; Distribution of the part due to activation and the part due to the contamination of the structures and components;Mapping of the dose equivalent rates in the reactor block and their evolution in time, under the specific influence of the different sources. To meet these objectives, the radiological inventory of the installation applies various methods, especially theoretical ones, using radiation protection calculation codes. The purpose of this paper is to describe the methods of radiological characterization in readiness for dismantling work and to organize the management of the waste produced by Creys-Malville reactor block dismantling operations.

1. INTRODUCTION

The evaluation of NPP structure and component activity has to take account of the methods used to declare the waste in relation to the different management channels. Therefore, it should take into account the two following components:

- Contamination;
- Activation.

2. ACTIVATION

2.1. Primary system

The flows of neutrons generated in the core during operation of the NPP, leads to activating nuclear reactions in the structures subjected to neutron irradiation.

The activity induced by the activation of the reactor block internal structures and components is essential data for defining the orientation of waste to the various storage channels and to ensure the radiological risk is controlled. The same problem occurs for each shut down NPP [1, 2].

As part of the studies performed for decommissioning the Creys-Malville Super-Phénix NPP, the activity calculations for the structures and primary system components break down into three stages:

- Calculation of the intensity of the sources in the fuel and fertile media at 320 JEPN and of the neutron flux inside the core;
- Calculation of the flux propagated outside the core based on the previously calculated neutron sources. The geometry used is cylindrical, two-dimensional of type R-Z, extending from the center of the core to the concrete reactor block, from the center of the core to the vessel head plug, and from the center of the core to the under-vessel concrete;

— Determination of the spot concentrations of activity in the primary system structures and components based on the fluxes obtained (per space interval and energy group), characteristics of activation reactions (microscopic thermal cross-section, resonance integral, fast macroscopic cross-section, energy threshold and so on), and the chemical composition of the materials.

Calculations of the neutron flux in the core and propagation beyond the core takes place using the ERANOS code, which is dedicated to the study of fast breeder reactors.

Activity calculations were carried out by taking account the past history of plant operation from its divergence to its shutdown on 24 December 1996, that is to say a total period of operation equivalent to 320 JEPN.

The activation calculations have shown that the highest activities are located in the lower axial zone below the core. The side and upper zones are protected better due to the presence of lateral and upper neutronic protections. Moreover, high cobalt content stellite coatings are encountered in the lower axial zone on the following components:

— Assembly supports;
— Truck/truck support contact surfaces.

The main source of gamma radiation during the first 20 years is ^{60}Co, the origin of which is the neutronic irradiation of natural cobalt, exclusively made up of ^{59}Co present in alloys and steels, by the capture of one neutron: ^{59}Co (n, γ)^{60}Co. The ^{60}Co is β-radioactive, with a radioactive half-life of 5.27 years, emitting two γ photons with 1.33 MeV and 1.17 MeV energy on each disintegration. Figure 1 shows the values for the maximum specific activity of the main components and structures of the reactor block ten years after plant shutdown (apart from the contamination source term).

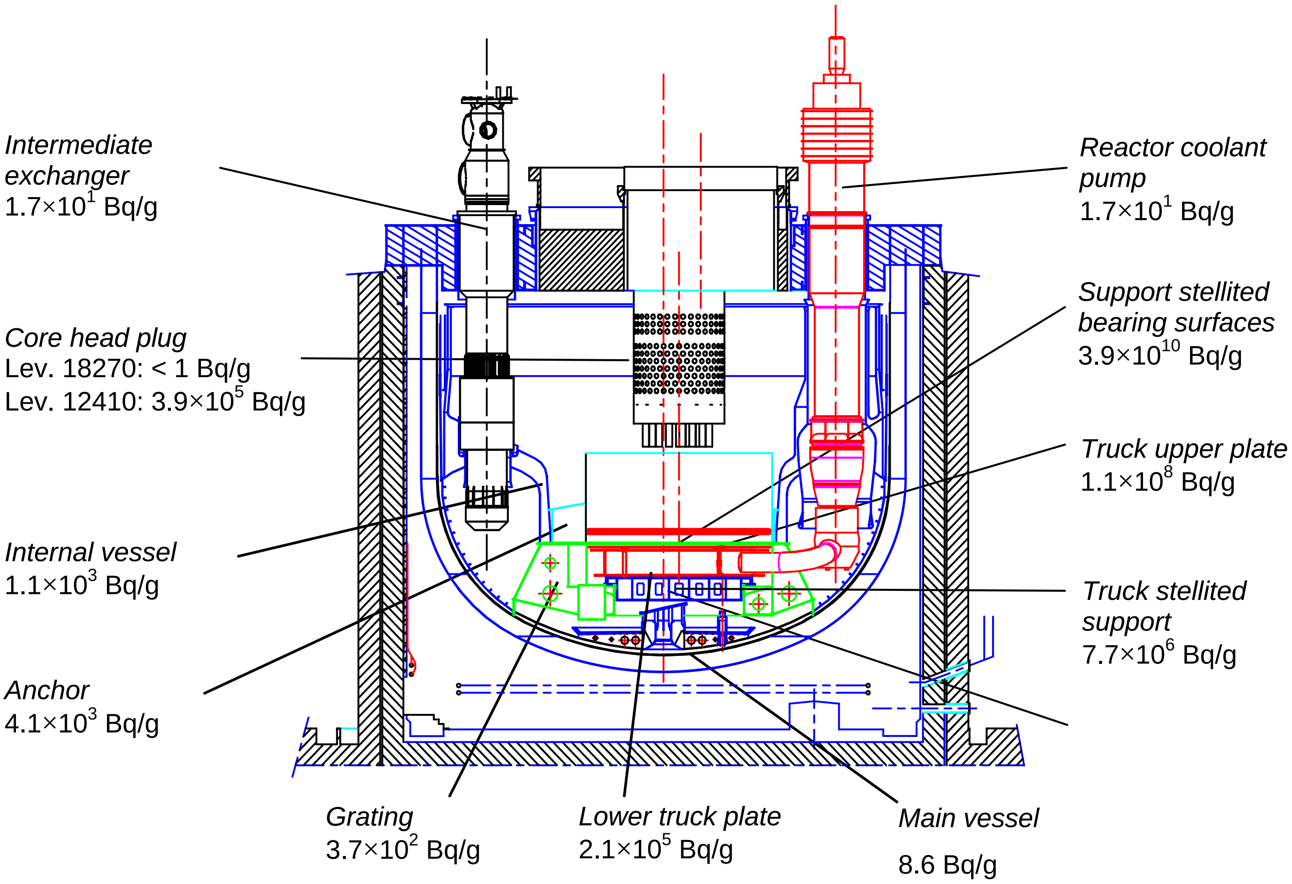

FIG. 1. Maximum activity concentration in Bq/g 10 years after shutdown.

In view of the large number of events that occurred during the life of the installation that could have consequences for the final radiological state (power modifications, operating incidents, etc.), the complexity of the calculations and the lack of details as to the source data such as the contents of impure elements in the materials, the uncertainty as to the results of activity is often considerable and difficult to quantify. It is impossible to have exact knowledge of the radiological condition through calculations alone. Consequently, these results have to be cross-checked as far as possible with readings obtained in the reactor block or on extracted components such as dose rates, samples, etc.

2.2. Secondary system

Taking into account their physical separation and the protections implemented in the primary system, the secondary systems are not subjected to the neutron flux and are not activated.

3. CONTAMINATION

The activity in the nuclear zones results mainly from contamination (the neutron fluxes were too low in normal operating conditions to activate the structures). This is caused by deposits of radio-elements present in the cooling liquid (primary and secondary sodium). Calculation of this component relies therefore on the creation of a list of radio-elements that are or could be present in significant quantities in the sodium.

3.1. Primary system

In the absence of a failed fuel element, the radionuclides present in the reactor coolant sodium may be derived from:

- The sodium itself;
- The impurities present in the supplied sodium;
- Fuel dust resulting from the production of pins on the outside of the sheaths.

Metal elements from the structures and components.

When these elements, which are subjected to neutron flux, present in the reactor block are activated, radioisotopes are created. The evaluation of the concentrations is based on the exact knowledge of the flux outside the core and the coefficients of dissolution/abrasion of metals in the sodium, as well as the consequences for their physicochemical behaviour. Certain radionuclides have physical and chemical behaviour which favors depositing or adsorption on the walls.

These assumptions are taken into account in order to obtain a coherent estimate of the sodium/wall distribution using all the information available. The approach adopted for the evaluation of the radioactivity of the sodium impurities and the elements coming from the dissolution of structure materials is as follows:

- Search for the sodium impurities that may be activated starting from the results of the analyses on the delivered sodium;
- Search for the elements of the materials of the structures placed in sodium solution which may be activated;
- Classification into families according to the probability of being rendered radioactive and according to the cross section and the concentration;
- Irradiation of the radionuclides during 320 EFPD by taking account of the past history of plant operation, and by taking the flux at the center of the core in order to calculate the activation of the primary sodium impurities.

The estimate of the activity of the fission products and the heavy radionuclides emitting alpha, beta and gamma radiation takes place by considering oxide pollution on the fuel and fertile pins on completion of their fabrication (a few heavy metal µg per pin).

The last stage consisted in comparing the activities of the radionuclides measured in the TASTENA (sodium samples) with the calculated activities and, by applying selection criteria, retaining the value which may be considered a ‘reference’ value in order to draw up the final list.

3.2. Special case of tritium

Tritium is also present in the sodium reactor coolant in significant quantities, the sources of tritium are:

- Ternary fission in the fuel;
- Nuclear reactions on the control rod boron;
- Activation of the boron and lithium impurities present in the delivered sodium, the fuel and the materials and structures;
- Reactions of the fuel pins with the pressurization helium.

The two first methods of formation are the most common. However, differences of concentrations allow the tritium to migrate in the opposite direction to hydrogen through the metal walls. This makes the theoretical evaluation of the radioactivity of this isotope difficult to assess. The value associated with tritium then results from measurements.

In view of the time that has passed since the shutdown of the Creys-Malville reactor and the evacuation of waste resulting from its dismantling, reactions with products having a period of less than 50 days were excluded from the search for possible activation reactions apart from the Ar37 referred to in the ANDRA specifications relating to the Aube surface disposal. In addition to the intrinsic sodium activity, the main elements contributing to the activity of primary sodium on 1st January 2001 are:

- Tritium;
- Corrosion products: ^{55}Fe, ^{59}Ni, ^{63}Ni, ^{57}Co, ^{60}Co, ^{54}Mn, ^{65}Zn...;
- Fission products: ^{137}Cs, ^{99}Tc, ^{93m}Nb, ^{93}Mo, ^{133}Ba, ^{106}Ru...

3.3. Secondary sodium

The activity of the radio-elements present in the secondary sodium was determined by considering the radiation, during 320 EFPD, of the sodium and metal element impurities resulting from the solubilization of structure materials under the maximum flux observed at the intermediate exchangers. Following the calculations, it appears that the activity of the secondary sodium is primarily due to the tritium produced in the reactor coolant circuit and which migrates by diffusion through the metal walls of the intermediate exchangers. This radioactivity was estimated from the tritium measurements made on the TASTENA’s samples.

3.4. Procedure for sodium characterization

The sodium is hydrolysed under moist air atmosphere with argon as sparging gas (Fig. 2). The aqueous phase is analyzed.

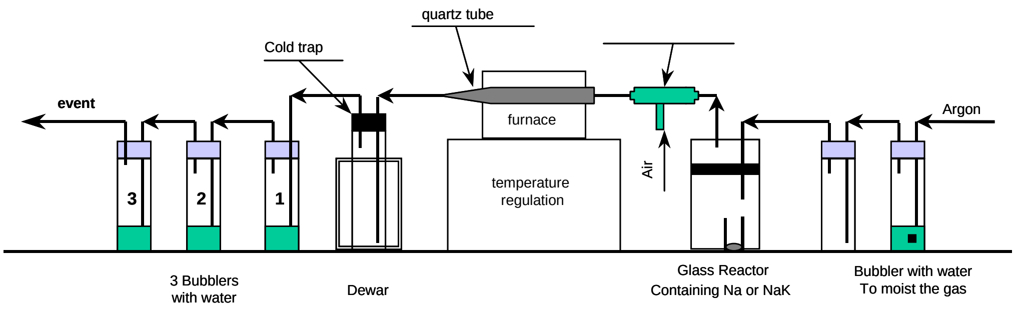

FIG. 2. Scheme of sodium hydrolyse.

3.4.1. Tritium

Generated water trapped in dewar and in bubblers are analyzed by liquid scintillation counting (detection limit, DL: 50 Bq/L) after addition of liquid scintillation cocktail. After distillation of soda solution, the distillate is measured too. (Accuracy is 10%).

3.4.2. Activation products

- ^{14}C: Production of CO_2 by oxidation (several ways can be used). CO_2 is trapped in a basic solution or specific reagent (carbosorb) and measured by liquid scintillation (DL: 0.2 Bq/g Na). (Accuracy is 10%);

- ^{36}Cl: After addition of stable Cl as carrier, the Cl is precipitated with Ag_2NO_3. The precipitate is solubilized with sodium thiosulfate. ^{36}Cl is measured with liquid scintillation and the yield with ionic chromatography. (DL: 3 Bq/g Na). (Accuracy is 10%);

3.4.3. Corrosion products

- ^{55}Fe: the solution is acidified and evaporated to eliminate tritium. Natural iron is added as carrier. After acid solution and chemical liquid-liquid extraction with MIBK (methylisobuthylceton) to isolate Fe, the solution is counted with liquid scintillation (DL: 10 Bq/g Na). The chemical yield is measured with atomic absorption spectrometry (AAS). (Accuracy is 10%);

- ^{63}Ni: Natural nickel is added as carrier. Dimethyl-glyoxime (DMG) is used to extract nickel with chloroform as solvent. Chemical yield is measured by AAS and the solution counted with liquid scintillation (DL: 5 Bq/g Na). (Accuracy is 10%);

- ^{57}Co, ^{60}Co, ^{54}Mn, ^{65}Zn, are gamma emitters and measured with gamma spectrometry (DL < 1 Bq/g Na).

3.4.4. Fission products

- ^{134}Cs, ^{137}Cs, ^{133}Ba, ^{93m}Nb, ^{106}Ru… are gamma emitters and measured with gamma spectrometry. (DL < 1 Bq/g Na);

- ^{90}Sr: A pre-treatment of sample with co-precipitation with lanthane hydroxide remove 99.9% of ^{22}Na which interferes with ^{90}Sr. Otherwise liquid-liquid extraction with crown ether, the solution is measured with liquid scintillation. (DL: 3.5 Bq/g Na). (Accuracy is 10%);

- ^{129}I: The iodine is oxidized with $NaNO_2$ to generate I_2. After the extraction with $CHCl_4$ or CCl_4 and a back extraction with sodium sulfite (Na_2SO_3) addition, ^{129}I is measured with low energy gamma spectrometry. The chemical yield is measured with ICPMS or with ^{131}I spike. (DL: 0.04 Bq/g Na). (Accuracy is 10%).

3.4.5. α emitters heavy nuclei

^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am: After addition with a ^{242}Pu or ^{243}Am tracer, an oxido-reduction phase is necessary for the extraction. A specific separation on 'TRU' resin and an electrodeposit on a support in thin layer lead to the measurement with alpha spectrometry (DL: ~ 0.1 Bq/g de Na for each isotopes). (Accuracy is 10%).

3.4.6. $\beta\gamma$ emitters heavy nuclei

The ^{241}Pu: addition of a second tracer before electrodeposition to calculate the lost of the deposit. The solution is measured with liquid scintillation. (DL: 2.5 Bq/g de Na). (Accuracy is 10%).

4. DOSE RATES GENERATED BY THE ACTIVATION OF STRUCTURES AND PRIMARY COMPONENTS

In order to ensure that the radiological risk is controlled during the removal and treatment operations and to meet the requirements for waste management (channels, conditioning, etc.), the dose rates around the components, in the reactor block, on the slab, in the reactor cavity and in certain specific zones of the core support structures involved in the retention tank resorption operations, are evaluated. The dose equivalent rates at different points of the reactor block will be used to optimize the decommissioning strategy and the dismantling operations to be defined: remote operations, contact interventions, definition of the biological protections and so on.

The calculations carried out are performed for several advanced states of the plant decommissioning operations. Based on the γ radiation spectra determined from the isotope activities for the selected cooling times, the dose equivalent rates are mapped using the MERCURY 5 computer code. This software covers three-dimensional geometries and integrates the straight line spot attenuation nuclei for gamma radiation (biological dose, heating, etc.) using a Monte Carlo technique in the multi-group approximation. As an example, Fig. 3 shows the dose equivalent values obtained at various points of the Super-Phénix reactor block generated by the activation of the structures and the primary system components at 1st January 2010 in the drained vessel configuration.

The dose equivalent rates in the reactor cavity and the reactor block are essentially due to the structures located in the lower axial zone below the core (truck and/or truck support). These structures, which include the stellite coatings with high cobalt content (assembly supports, truck/truck support bearing surface), contribute to more than 99% of the total dose equivalent rate.

5. CONCLUSION

Decommissioning operations will generate waste requiring a general approach to site management and preparations for their evacuation towards the various suitable channels.

In order to attain this goal, the first step is the complete radiological appraisal of the installation. In this context, a quantitative and qualitative evaluation of the primary and secondary systems was carried out.

The information thus obtained will then enable:

- Analysis of the destination of the various wastes and effluents generated by the decommissioning of the radioactive and/or contaminated systems having been in contact with the primary or secondary sodium and/or its aerosols;
- Definition of the elements needed to establish the principles of treatment, management and the destination of this waste from the places of production up to the stacking areas for evacuation;
- Definition of the elements needed to establish the statutory documents for their evacuation from the site.

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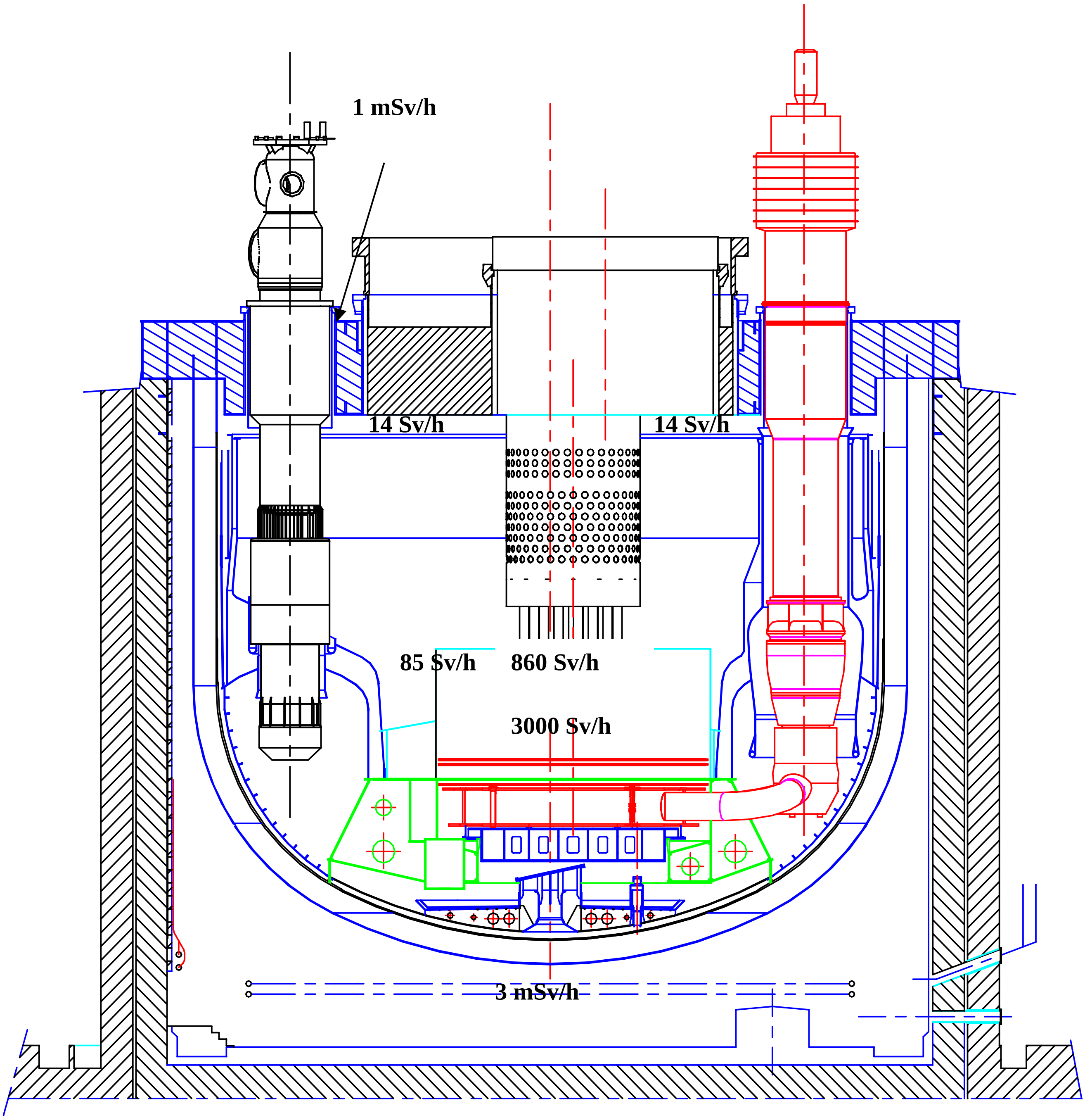

FIG. 3. Dose equivalent rate in reactor block vessel drained on 1st January 2010.

FRENCH SODIUM WASTE STORAGE RULES

F. BAQUÉ, K. LIGER, G. RODRIGUEZ
COMMISSARIAT A L'ENERGIE ATOMIQUE (CEA), CADARACHE,
SAINT PAUL LEZ DURANCE, FRANCE

TH. BOUCHERAT, J. OLLIVIER, E. JOULIA
ELECTRICITE DE FRANCE (EDF) – CIDEN, VILLEURBANNE, FRANCE

Abstract

In the frame of Superphénix Plant decommissioning, CEA and EDF had to determine the rules to apply for safe sodium waste storage. Even if sodium waste storage has been monitored for some decades (but only during Operational Plant phases), some recent events showed that this item had to be secured before beginning large decommissioning operations. Of course, the best way would be an on-line treatment but operational constraints always imply a delay in this operation. Indeed, a number of sodium wastes will be produced during the period before the end of Superphénix sodium treatment (planned in 2013) and will have to wait for further treatment. The events to be avoided, or at least taken into account, are uncontrolled sodium reaction with air moisture (large hydrogen production, important overheating) and sodium reaction with liquid water (pressure waves, large hydrogen production, important overheating). Careful analysis of all abnormal events in sodium waste storage disposal was performed and led to rule evolution. In 2004, experimental studies were undertaken, in order to know how solid sodium at room temperature reacts with air humidity: the conditions of aqueous sodium hydroxide production (which is the main risk source in sodium waste storage) have been observed. Waste classifying: pure sodium and soda to be separated, bulk and residues to be separated; Sodium waste containers: tight, dry, easy to refill with gas, protected against overpressure effect, with specific marking and reference; Dedicated rooms: dry, with specific markings, with specific sodium fire extinguishers; Maximum duration: three months before next refill with inert dry gas, in an over-container if more than one year; Dry gas feeling: inert gas except for sodium film residues (dry air). For Superphénix application, packaging and storage conditions of sodium wastes have been defined, in accordance with container fluxes to sodium waste treatment cell: it was decided to initially fill the containers (packaging phase) with dry air (dew point less than -10°C) whatever are the sodium waste types, because any gas in the container is rapidly ant totally dried by sodium itself. Then, renewal of gas in the containers will be done with dry argon (storage phase), except for sodium film residues (dry air filling). On site feedback experience will confirm the efficiency of these recent rules which have to be adapted to each specific case (sodium waste type, containers) and which can evolve with on site feedback experience.

1. INTRODUCTION

The duration of the decommissioning of a Liquid Metal Fast Reactor (LMFR) is estimated to several years and in function of the size and the complexity of the reactor to be decommissioned, it can be extended to several decades.

The liquid metal fast breeder reactor Superphénix is being dismantled since 1998. The sodium coolant has to be removed out of primary and secondary circuits and components have to be cut and treated in order to eliminate all remaining sodium traces, films and accumulations, before coming to waste disposal plant. The treatment of these components is based on mechanical cutting in air at room conditions, then cleaning for conditioning in adapted containers. Hence, the safety approach of the cutting operations has to be clearly defined.

Hence, this paper proposes to remind the chemical reaction between metallic sodium and air, their consequences and the influence of moisture in air. A thermodynamic approach of the equilibrium between all the chemical reactions that can be encountered is first presented. Then a presentation of the works done to evaluate the influence of moisture of air in contact with sodium, and the associated kinetics will allow, in the future, concluding to some general recommendations for the cutting operations of components.

In the frame of Superphénix Plant decommissioning, CEA and EDF had to determine the rules to apply for safe sodium waste storage. Even if sodium waste storage has been monitored for some decades (but only during Operational Plant phases), some recent events showed that this item had to be secured before beginning large decommissioning operations. Of course, the best way would be an on-line treatment but operational constraints always imply a delay in this operation.

Indeed, a number of sodium wastes will be produced during the period before the end of Superphénix sodium treatment (planned in 2013) and will have to wait for further treatment.

The events to be avoided, or at least taken into account, are uncontrolled sodium reaction with air moisture (large hydrogen production, important overheating) and sodium reaction with liquid water (pressure waves, large hydrogen production, important overheating). Careful analysis of all abnormal events in sodium waste storage disposal was performed and led to rule evolution.

In 2004, experimental studies were undertaken, in order to know how solid sodium at room temperature reacts with air humidity: the conditions of aqueous sodium hydroxide production (which is the main risk source in sodium waste storage) have been observed. The analysis of these conditions helped EDF and CEA to define the right conditions for packaging and storage of sodium waste.

2. CHEMICAL PROPERTIES OF METALLIC SODIUM

2.1. Generality [1- 3]

Sodium belongs to the alkali metal family and is therefore a very strong reducer. At room temperature this product is solid. Due to its reactivity with oxygen and water, pure sodium can be kept in that state only under pure inert gas such as argon or nitrogen. In laboratory it is very common to find it immersed in oil or petrol.

Metallic sodium is commonly used in chemical industry as a reducing reagent or as a catalyst. In nuclear field it has been widely used as coolant of Liquid Metal Fast Reactor because of its exceptional physical properties: low density, good heat conductivity, liquid over a wide range of temperature (from 97.8°C until 883°C), compatible with fast neutron spectrum, no toxic and very important availability on Earth (as sodium chloride in the oceans). The main drawback of the use of sodium is that it can react violently with water and actively with air especially when sodium is at liquid state. Indeed, liquid sodium can ignite easily in air or oxygen, producing sodium oxide and super oxide fumes and a typical orange light. Hence, at room temperature, solid metallic sodium will be oxidized in contact with air.

2.2. General description of the reaction of solid metallic sodium with air at room temperature [4]

Color of pure solid sodium is near to the color of mercury: metallic silver. But in contact with air this color will quickly change to gray (in few seconds). This change of color is the proof that there is a chemical reaction at the surface of the sodium. This chemical reaction is due to a surface oxidation, transformation of metallic sodium to sodium oxide in contact with oxygen:

$$2\mathrm{Na}\,(\mathrm{s}) + \frac{1}{2}O_2(\mathrm{g}) \rightarrow \mathrm{Na_2O}(s) \qquad \Delta H_r = -141\ \mathrm{kJ/mol} \qquad (1)$$

At higher temperature (250-300°C), the major compound produced by oxidation is Na_2O_2 but at room temperature, only Na_2O is produced. With time, several minutes later, the appearance of the surface of the solid sodium turns to white and sometimes a thin layer of liquid product appears. It is important to say that the evolution of sodium under air strongly depends on the weather (dry or wet air) and the same basic experiment can turn to different results if the atmosphere conditions are not constant. This observation indicates that several chemical reactions can occur to the surface of this solid sodium in contact with air. In fact sodium reacts with the oxygen of air but can also react with the moisture present in air.

$$\mathrm{Na} + \mathrm{H_2O(g)} \rightarrow \mathrm{NaOH} + \tfrac{1}{2}\,\mathrm{H_2} \qquad \Delta H_r = -151\ \mathrm{kJ/mol} \qquad (2)$$

Moreover the products created by the reaction with oxygen or moisture can also react with moisture/air to generate other products: transformation of sodium oxide to sodium hydroxide, hydration of sodium hydroxide, carbonation of sodium hydroxide to sodium carbonate. Though the low carbon dioxide concentration in air (around 300 ppm), carbon dioxide is also participating to the chemical reaction. Thus, the following chemical reactions are also observed with time:

$Na_2O(s) + H_2O(g) \rightarrow 2\ NaOH(s)$ $H_r = –$ kJ/mol (3)

$NaOH(s) + n\ H_2O\ (g) \rightarrow NaOH, n\ H_2O$ $n = 1,2,3,4$ or 5 $H_r =$ depends on n (4)

$H_r =$ nearly – 20 kJ/mol

$2\ NaOH(s) + CO_2(g) \rightarrow Na_2CO_3(s) + H_2O(g)$ $H_r = -123$ kJ/mol (5)

$Na_2CO_3(s) + CO_2(g) + H_2O(g) \rightarrow 2\ NaHCO_3(s)$ $H_r = -35$ kJ/mol (6)

Sodium carbonates are solid friable compounds which can form a thick layer on the surface of the metallic sodium. This solid layer of sodium carbonate could lower the evolution of transformation of the sodium with air or even prevent it from occurring. Thus it has been observed some metallic sodium covered with sodium carbonate under air for several years with no apparent evolution. Reaction occurs only under excess of CO2. Anyway, carbonation of sodium occurs very slowly under dry air conditions, due to the low concentration of CO2 in air. In presence of wet air, this reaction is not seen as sodium hydroxide is more eager to react with water than CO2. Hence, for cutting operations which last about 3 months maximum, reaction between solid metallic sodium and carbon dioxide can be neglected, and only reactions between Na, moisture and oxygen have to be taken into account. Anyway, for storage studies, reactions with CO2 should have to be taken into account.

The purpose of safety recommendations during cutting operations is to avoid sudden reactions due to:

- Violent reaction between metallic sodium and hydrated soda;
- Hydrogen ignition and explosion.

The number of chemical reactions involved at the same time, and the competitiveness of these chemical reactions between each other lead us to the following conclusions:

- It is difficult to make appropriate recommendations on the safety of component cutting operations with regard of air composition;
- It is necessary to perform chemical experiments of solid sodium under selected air composition to estimate the effect of moist air on metallic sodium.

Thus the studies done to solve this problem were based on two major objectives:

- A bibliographical research of all the thermodynamic equilibrium between all the components that can be encountered when solid sodium is in presence of wet air,
- An experimental study on the evolution of solid sodium under controlled air.

3. THERMODYNAMIC EQUILIBRIUM OF THE COMPONENTS PRODUCED BY THE REACTION OF SOLID SODIUM WITH HUMID AIR

3.1. Definition of the different products and chemical reactions [5, 6]

Considering the low concentration of CO_2 in air and the time scale of this study, reaction involving CO_2 are not considered.

3.1.1. Sodium and water

Sodium reacts with excess of water according to the following chemical reaction. In normal conditions, this reaction is rapid and exothermic

$Na(s) + H_2O(l) \rightarrow NaOH(s) + ½ H_2(g)$ $H_r = -151$ kJ/mol (2)

At high temperature and with an excess of sodium and with water vapor, the chemical reaction between sodium and water can lead to the formation of sodium oxide and sodium hydride.

$4 Na(s) + H_2O(g) \rightarrow Na_2O(s) + 2 NaH(s)$ (3)

3.1.2. Sodium oxide and water

Sodium oxide is obtained from the sodium oxidation with air. With water, sodium oxide is transformed to sodium hydroxide according to the following chemical reaction:

$Na_2O(s) + H_2O(g)\rightarrow 2 NaOH(s)$

3.1.3. Sodium oxide and hydrogen

At room temperature hydrogen does not react with sodium, but between 200°C and 350°C, sodium hydride can be formed.

$Na_2O(s) + H_2(g)\rightarrow NaOH(s) + NaH(s)$

In the scope of this study, production of hydrogen is low (large amount of air and few amount of Na). Furthermore, temperature for cutting operation remains low. Hence, this reaction can be neglected.

3.2. Stability of the different products – Stability diagrams

Thermodynamical study was done in order to estimate which reaction is more stable and which products are likely to be observed. Anyway, the results obtained should be taken into account with care as kinetics aspects are neglected.

This study consists in listing all reactions possible between Na and H_2O considering all possible products (NaOH, NaH, Na_2O). Then, by the mean of HSC Chemistry© computing code [7], it was possible to define the Ellingham diagrams (variation of free enthalpy $\Delta_rG°$) as a function of temperature, to evaluate the reaction of sodium with water vapor.

The following chemical reactions were taken into consideration:

$Na(s) + H_2O(g) \rightarrow NaOH(s) + ½ H_2$

$2 Na(s) + H_2O \rightarrow NaOH(s) + NaH(s)$

$2 Na(s) + H_2O(g) \rightarrow Na_2O(s) + H_2$

$4 Na(s) + H_2O(g) \rightarrow Na_2O(s) + 2NaH(s)$

The stability diagrams can be represented in function of one mole of sodium (case of an excess of water), see Fig. 1, or in function of one mole of water (case of an excess of sodium), see Fig. 2.

From these figures it can be deduced that an excess of water will favor the creation of sodium hydroxide and hydrogen. On the opposite, an excess of sodium will favor the creation of sodium oxide and sodium hydride. This result is in total accordance with that can be found in bibliographical data. But this conclusion is only partially solving the problem of metallic sodium in contact with air because:

- The stability diagram does not consider the kinetics of the different chemical reaction;
- It is not obvious that at the sodium surface there is an excess of sodium or an excess of water: it mainly depends of the water vapor concentration in air.

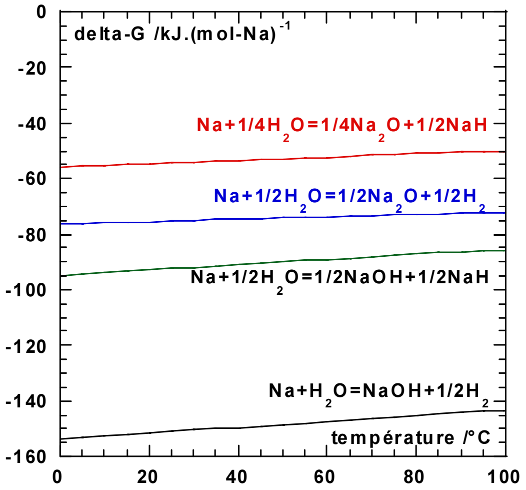

FIG. 1. The stability diagram of sodium oxidation with water vapor (per mole of sodium).

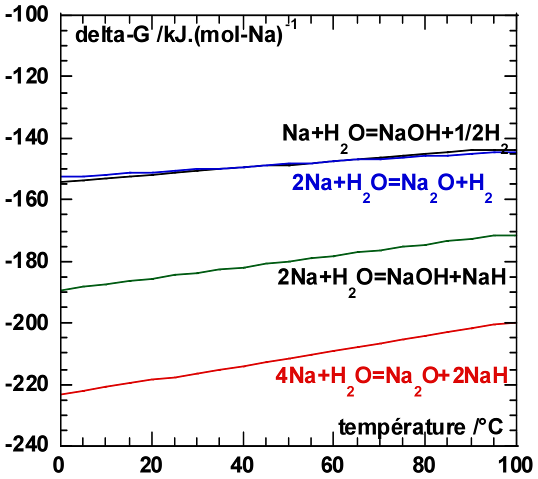

FIG. 2. The stability diagram of sodium oxidation with water vapor (per mole of water).

4. EXPERIMENTS ON THE EVOLUTION OF SOLID SODIUM UNDER CONTROLLED AIR

The previous paragraph has presented the thermodynamic aspect of the chemistry of sodium in presence with moisture at room temperature. But, important parameters like kinetics aspects and surface interaction, which were not taken into account in the first part of this paper, have to be investigated further in order to estimate the behaviour with time of solid sodium under air and the influence of water concentration in air.

To prevent accident when manipulating metallic sodium under atmospheric conditions, several barriers should be hold:

- Maintaining low humidity to prevent formation of sodium hydroxide which presents great affinity for water and could generate sodium/water reactions by contact of hydrated soda with residual Na;
- Minimizing the presence of metallic sodium bulk and a least avoid bulk on horizontal areas;
- Avoiding non venting zone where hydrogen could accumulate.

For the first barrier, it is necessary to know the maximal limit of humidity under which no aqueous sodium hydroxide appears during the whole treatment duration. Furthermore, this limit should be determined with respect to temperature, as this parameter is assumed to have an effect on reaction kinetics.

In Superphénix, the treatment of primary components is based on mechanical cutting in air at room conditions [8], then cleaning for conditioning in adapted containers. Existing cleaning pits of the plant will be used. The component treatment is as follows: each component is removed from primary liquid sodium at 180°C, handled in a specific basket under inert gas, then cooled in inerted pit before transfer in air to cutting workshop inside the reactor building itself. After visual checking, accessible remaining sodium is removed by hand. Cutting is finally performed to generate standardized little stainless steel pieces for further water cleaning. All the operations performed in air conditions have to be secured and violent reaction as well as hydrogen inflammation avoided. Concerning hydrogen inflammation, renewable of air is assured to remain hydrogen concentration low enough to prevent inflammation and explosion.

4.1. Description of the tests

Two types of tests were performed to represent the 2 types of dismantling operations:

- Cutting of small components containing sodium films.
- Cutting of large components containing sodium films and bulk metallic sodium.

The purpose of the tests performed was to determine the maximum humidity under which no aqueous soda is produced during the whole duration of the treatment, considering that the small components cutting operation duration is about 2 weeks and the large components cutting operation duration is about 3 months. The general objective is to find a composition of wet air for which cutting operation is safe (without production of hydrated soda).

4.1.1. Small components case

Cutting operation is done under air in a workshop inside the reactor building itself. The treatment duration of one component is less than 2 weeks and the atmosphere of the reactor building contains between nearly 3 g H_2O/kg air and 10 g H_2O/kg air. To simplify the cutting operation of the component, it is important to determine the maximum humidity under which no aqueous soda is produced during the whole duration of the treatment. (i.e. 2 weeks). The residues of cutting contain only metallic sodium thin layers (thickness of 3 mm maximum).

Tests consist in observing the surface behaviour of sodium metallic films under controlled humid air in order to detect the beginning of formation of aqueous soda. Analysis of the gas atmosphere was also performed to verify the atmosphere composition and three identical samples are tested simultaneously for reproducibility assessment. The parameters to be taken into account are:

- Temperature of the test;
- Humidity of air (between 3 g H_2O/kg air and 10 g H_2O/kg air);
- Atmosphere renewable. In the cutting workshop, the volumetric exchange rate of air is between 1 and 10 volume of the building per hour. Considering the volume of the experimental glove box in which experiments were done and the maximum range of the mass flow controller, a volumetric exchange rate of 1 volume per hour was chosen. This is underestimated compared the real conditions in the cutting workshop. Hence, it will be necessary to verify that the concentration of water is in large excess.

During the experiments, samples were maintained vertically to avoid formation and stagnation of aqueous soda on horizontal surface containing metallic sodium, which could generate unexpected water/sodium reaction. But, during cutting operation, components can be maintained either horizontally or vertically. To find the humidity limit, 'dichotomy method' was used. It consists in performing tests under the minimum and maximum water concentrations and then in testing the middle point of water concentration. This leads to identify the range of humidity in which stands the transition point, which we are looking for. Hence, the following table gives an example of tests matrix:

4.1.2. Large components case

Large components contain thin layer of metallic sodium and in some case, bulk metallic sodium. The duration of the cutting treatment of large components is approximately 3 months. In order to simplify the cutting operations, it is interesting to work under atmospheric conditions while avoiding the production of aqueous soda. Two solutions are possible:

- The first one consists in working under reactor building atmosphere using an inert protective shield, in which the component is placed. Only the part under cutting is put in contact with the workshop atmosphere. This strategy has many constraints and is not easy to manage.
- The second solution consists in drying the workshop atmosphere to a humidity level compatible with 3 months of working under air conditions without production of aqueous soda.

Hence, the purpose of second series of tests was to find the humidity level to reach for safe cutting operation under controlled atmosphere condition. These tests were performed on thin layer and bulk metallic sodium. Considering that this test should last for 3 months, only one humidity value chosen on behalf of the results of small components tests was selected.

TABLE 1. TESTS MATRIX FOR SMALL COMPONENT CUTTING FIELD

Test n°	Atmospheric conditions	Pressure, temperature Flow rate	Geometry studied
1 (3 simultaneous tests)	Air + 10g H_2O/kg air (1.6%V of water)	Atmospheric pressure, 35°C 500 L/h volumetric exchange rate: 1	Thin layer of metallic sodium (thickness 2 mm) on vertical stainless steel plate (2*20 cm)
2 (3 simultaneous tests)	Air + 3 g H_2O/kg air (0.48 %V of water)	Atmospheric pressure, 35°C 500 L/h volumetric exchange rate: 1	Thin layer of metallic sodium (thickness 2 mm) on vertical stainless steel plate (2*20 cm)
3 (3 simultaneous tests)	Air + 7 g H_2O/kg air (1.0%V of water)	Atmospheric pressure, 35°C 500 L/h volumetric exchange rate: 1	Thin layer of metallic sodium (thickness 2 mm) on vertical stainless steel plate (2*20 cm)
N > 3 (3 simultaneous tests)	Air + x g H_2O/kg air *	Atmospheric pressure 35°C 500 L/h volumetric exchange rate: 1	Thin layer of metallic sodium (thickness 2 mm) on vertical stainless steel plate (2*20 cm)

4.2. Description of the test facility (CARNAC)

CARNAC facility (see Fig. 3) is composed of a series of glove boxes in which the atmosphere composition is controlled:

- 4 independent glove boxes (500 L each) in which reaction tests are done. Each glove box has its own regulated alimentation in gases (nitrogen, carbon dioxide, air and/or water) and its own temperature regulation from 25 to 70°C;
- 1 inerted glove box (700 L) for preparation of samples and test;
- 1 tunnel for the transfer from the preparation box to one of the testing glove boxes;
- 1 analysis system (gas micro chromatography and mass spectroscopy) in charge of the speciation of the atmosphere of glove boxes;
- 1 system for video acquisition and image treatment in order to observe the formation of aqueous soda. In addition, a dry towel impregnated with phenolphthalein is placed under the sample so that a fuchsia mark is observed when aqueous soda drop falls down.

4.3. Results of tests related to small components treatment

For each atmosphere condition tested, 3 stainless steel supports containing thin layer of metallic sodium were placed in the testing glove box and observed over the whole test duration (2 weeks). When aqueous soda is formed, bubbles appear on the surface and droplets fall down on the towel, generating a fuchsia mark (see Fig. 4). In some cases, projection of aqueous soda was observed before flowing of aqueous soda (see Fig. 5).

Summaries of the tests performed and specific remarks on each test are shown in Tables 2 and 3, respectively.

FIG. 3. CARNAC facility.

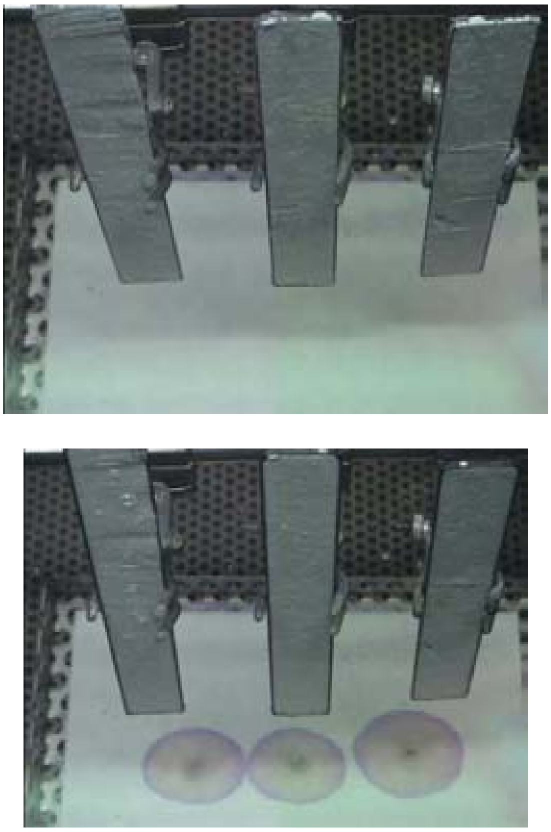

FIG. 4. Samples at the beginning and at the end of the small component tests.

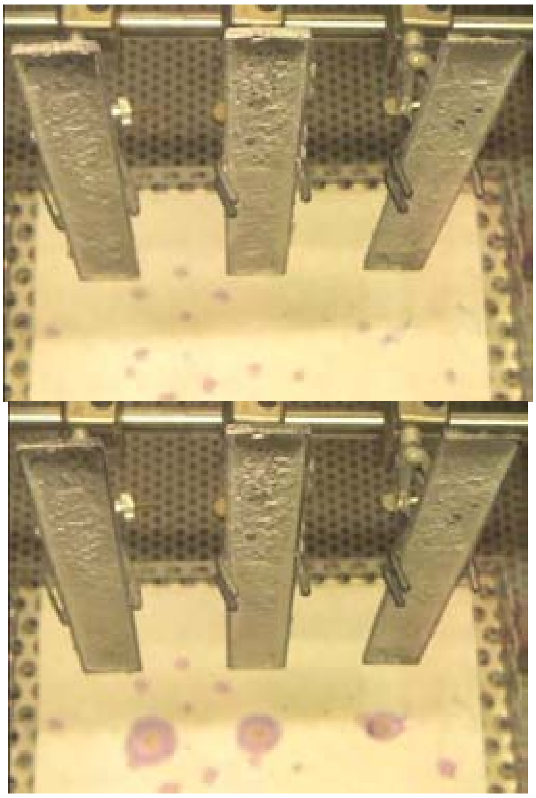

FIG. 5. Projection of aqueous soda during the small component tests.

TABLE 2. SMALL COMPONENT TEST EXPERIMENTAL RESULTS AT 35°C

Test N°	Operating conditions	Remarks
1	35°C, P = Atmospheric pressure Flow rate: 500 L/h 9.9 g of water/kg air (1.6%V)	Aqueous soda droplets after 4 hours of exposition No projection of aqueous soda before droplet flowing
2	35°C, P = Atmospheric pressure Flow rate: 500 L/h 3.1 g of water /kg air (0.5%V)	No aqueous soda over the whole test period (2 weeks)
3	35°C, P = Atmospheric pressure Flow rate: 500 L/h 6.2 g of water /kg air (1%V)	Aqueous soda droplets after 17 hours of exposition Projection of aqueous soda after 12 hours
4	35°C, P = Atmospheric pressure Flow rate: 500 L/h 5 g of water /kg air (0.8%V)	Aqueous soda droplets after 21 hours of exposition No projection of aqueous soda before droplet flowing
5	35°C, P = Atmospheric pressure Flow rate: 500 L/h 3.7 g of water /kg air (0.6%V)	No aqueous soda over the whole test period (2 weeks)

TABLE 3. SMALL COMPONENT TEST EXPERIMENTAL RESULTS AT 25°C

Test N°	Operating conditions	Remarks
6	25°C, P = Atmospheric pressure Flow rate: 500 L/h 3.7 g of water/kg air (0.6%V)	Aqueous soda droplets after 19 hours of exposition No projection of aqueous soda before droplet flowing
7	25°C, P = Atmospheric pressure Flow rate: 500 L/h 3.1 g of water/kg air (0.5%V)	Aqueous soda droplets after 21 hours of exposition No projection of aqueous soda before droplet flowing
8 Similar to test 7 but in addition to films, bulk sodium plate (thickness 2 cm) was placed in the glove box	25°C, P = Atmospheric pressure Flow rate: 500 L/h 3.1 g of water/kg air (0.5%V)	Aqueous soda droplets after: • 22 hours of exposition for thin films • 32 hours of exposition for bulk sodium No projection of aqueous soda before droplet flowing

Gas chromatography analysis of hydrogen in the gas atmosphere have shown that water was in large excess as hydrogen production represents consumption of less than 0.1% of water. For example, Fig. 6 shows a maximum hydrogen concentration in the gas phase of 500 ppmV, which corresponds to a consumption of 0.1% of water flow rate compared to the 1.6% available for reaction. Similar results were obtained for the tests at other water concentrations. Hence, a higher venting rate should note have any effect on the results obtained.

As water is in large excess, these results can be compared to former tests done under higher flowrate and using bulk metallic sodium in plates (see Table 4):

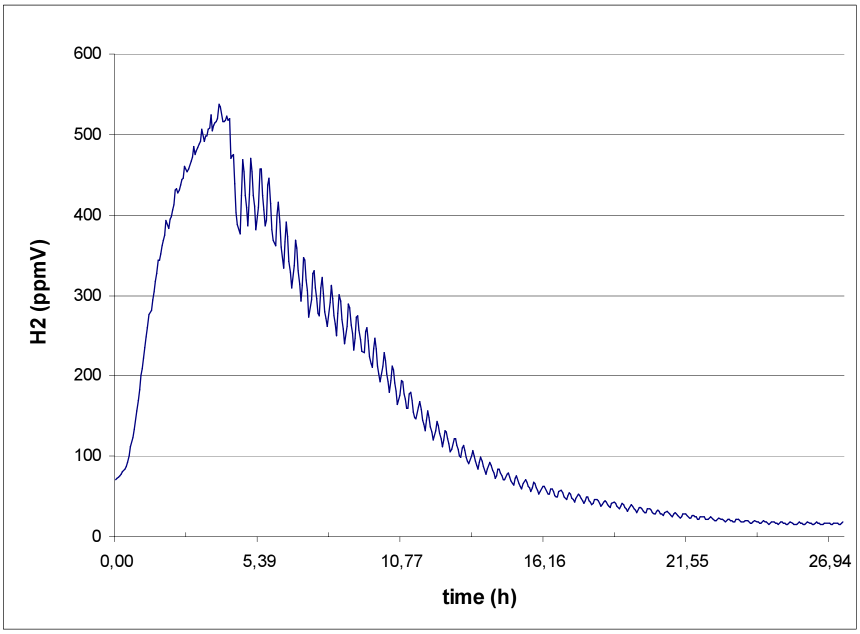

FIG. 6. Hydrogen concentration during test n° 1.

TABLE 4. FORMER EXPERIMENTAL RESULTS AT 20°C

Test n°	Experimental conditions	Remarks
9	20°C, P = Atmospheric pressure Flow rate: 10 m^3/h 4 g of water/kg air	Aqueous soda droplets after 24 hours of exposition
10	20°C, P = Atmospheric pressure Flow rate : 10 m^3/h 5.5 g of water/kg air	Aqueous soda droplets after 15 hours of exposition
11	20°C, P = Atmospheric pressure Flow rate: 10 m^3/h 7.7 g of water/kg air	Aqueous soda droplets after 6 hours of exposition

Figure 7 gives a representation of the tests results. Note that temperature has an effect on production of aqueous soda. The more humidity in air, quicker is the production of aqueous soda. It is difficult to find an explanation to these phenomena as no discrimination was done between complex phenomena occurring (mass transfer, kinetics, competition between oxidation and hydration). Anyway, more analytical tests are needed to find the theoretical explanation of the influence of temperature on the behaviour of metallic sodium under wet air. Furthermore, test n°8 shows that production of aqueous soda on bulk metallic sodium appears later that on metallic film. One explanation is that, on the thin films, aqueous soda appears after consumption of the whole layer of metallic sodium whereas on bulk metallic sodium, aqueous soda appears while amounts of metallic sodium still remains. Hence, aqueous soda can prematurely appear on the thin films. That is why tests 9, 10 and 11 on bulk metallic sodium have lead to late production of aqueous soda compared to other tests.

4.4. Test results for large components case

The purpose of these tests was to find a humidity limit below which no aqueous soda is produced over a period of 3 months. According to results on thin layer above, humidity was fixed below the limit found for small component case (i.e. below 3.7g /kg air). Hence, it was fixed at 3 g of water/kg of air. Only one test was performed at 35°C.

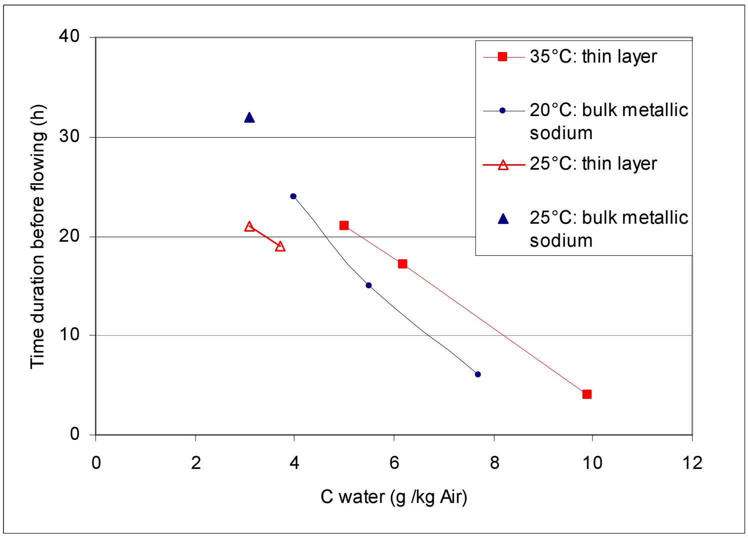

FIG. 7. Sodium behaviour under humid conditions.

TABLE 5. TESTS RESULTS FOR LARGE COMPONENTS CASE

Test N°	Operating conditions	Remarks
12 Thin layer of metallic sodium (thickness 2 mm) on vertical stainless steel plate (2 * 20cm)	35°C, P = Atmospheric pressure Flow rate: 500 L/h 3 g of water/kg air	Aqueous soda droplets after 20 days of exposition
13 Thin layer of metallic sodium (thickness 2 mm) on vertical stainless steel plate (2 * 20cm) Bulk metallic sodium on vertical plate (thickness 2 cm)	35°C, P = Atmospheric pressure Flow rate: 500 L/h 3 g of water/kg air	Aqueous soda droplets after: - 39.5 days of exposition for thin films - 40 days of exposition for bulk sodium

An early production of aqueous soda can be obtained when all metallic sodium is consumed (Test n°12 compared to test 13 has shown an early production of aqueous soda as all metallic sodium available for oxidation was consumed).

No production of aqueous soda occurs when metallic sodium remains available for oxidation. But, there is also a critical thickness of oxidized sodium (either Na_2O or $NaOH$). Below this critical limit (about 2-3 mm), diffusion of water and/or oxygen is possible and no aqueous soda is produced. But, above this limit, diffusion is too slow and aqueous soda is produced even if there is still metallic sodium available. This could lead to reaction between sodium and water associated to aqueous soda.

As a consequence, it seems difficult to guarantee that aqueous soda (i.e. no reaction between metallic sodium and water) will not occur over a period of 3 months as this phenomenon depends on several parameters like temperature, thickness of bulk metallic sodium and geometry of components.

In conclusion, secure cutting of large components with bulk metallic sodium can not be assured under atmospheric conditions except if humidity is removed.

FIG. 8. Test n°13 (large component case).

5. SUPERPHENIX APPLICATION

Superphénix decommissioning operations of sodium loops induce 4 types of sodium wastes:

- Metallic objects with sodium film: 30 microns thick sodium film on vertical surfaces and 1 mm on horizontal surfaces (for example: internal surfaces of sodium pipes);
- Metallic objects with bulk sodium: in areas difficult to drain or not drained (for example : blocked pipes, objects with angle shape, small diameter tubes);
- Residues of metallic sodium: small amounts of sodium which are removed during manual cleaning operations (before cutting operations of sodium components);
- Other wastes which were in contact with sodium: gloves, tissues, sponges which are used for sodium film and soda sweeping, plastic sheets used for floor protection. These wastes have no sodium but sodium oxides, hydroxides or carbonates.

5.1. Containers characteristics

Sodium waste containers are metallic and tight (no water ingress).

- Containers 8 m^3 (1.85×1.7×1.8 m internal) (5.2 tons of wastes) (3 mm removable roof) (joint ethylene propylene) (2 mm thick carbon steel wall);
- Drums 213 L (∅ 610 mm height 880 mm) (0.9 mm thick carbon steel) (removable roof with polymer joint and if necessary 2 valves for inerting and pressure measurement/safety valve);
- Stewpan 10 L (stainless steel) (∅ 268 mm) (removable cap with Nitril joint, safety valve 0.9 bar and if necessary 1 valve for inerting and pressure measurement).

5.2. Packaging rules

For all types of sodium wastes:

- Metallic, tight and closed container;
- Clean and dry container;
- Absorbing carpet on the container bottom;
- Filling only with dry sodium (if not, sweaping of sodium surface with specific tissue);
- No vinyl in the containers;
- When opening the containers, checking aqueous soda and sweaping if necessary;
- Shorten as much as possible the delay between sodium waste production and packaging;
- Metallic sodium waste objects have to be vertical inside the containers (to help aqueous phase drop) and sodium packs near the bottom;
- Container inerting: initial filling with dry air (dew point < -10°C);
- Container inerting: next fillings with ambient air (shorten the delay when filling the container with a new waste and assume container tightness between 2 filling operations.

5.3. Packaging rooms

- Clean, delimited, marked out room with few people coming inside;
- Risk signs at room gates and on containers: sodium, no water, inerting if necessary;
- Identification on the containers of quantity and type of sodium wastes;
- Protection against any external water ingress: rain, infiltrations, internal flooding;
- Smoke detector;
- Specific Marcalina powder extinguishers (only this type of extinguishers);
- Zoning: in case of radiological risk a sodium waste transfer, vinyl packing is necessary, but vinyl will be removed before sodium waste packaging.

5.4. Storage rules

5.4.1. General rules

Storage room:

- Clean, delimited, marked out room with few people coming inside;
- Risk signs at room gates and on containers: sodium, no water, inerting if necessary;
- Identification of all the containers of sodium wastes;
- Protection against any external water ingress: rain, infiltrations, internal flooding.
- No condensation on outer surface of containers;
- Smoke detectors;
- Specific Marcalina powder extinguishers (only this type of extinguishers).

5.4.2. Complementary rules

5.4.2.1. Objects with sodium film

Storage state: containers with ambient air inside, with absorbing carpet, with vertical objects, with tight caps.

5.4.2.2. Objects with bulk sodium

Storage state: containers with inert gas (argon) inside, with absorbing carpet, with vertical pipes, sodium packs near the bottom inerting renewing: it is based on a 3 month period but could be longer, depending on current feedback experience.

Over-containers could also be used, to store a number of containers and perform easily inerting renewal. Storage conditions for containers not coming from reactor block (always under inert gas): containers with caps equipped with specific device allowing inerting operations systematic inerting renewal of all containers, every 3 months all data being written on a specific form (container reference, inerting renewal date, observations…)

5.4.3.3. Objects with sodium residues

Container state during storage phase:

- In specific storage rooms (see hereafter);
- In the same conditions, whatever the container (stewpans) origins;
- Absorbing carpet on the container bottom.

For containers coming from reactor block:

- Rapid changing of cap: specific cap allowing inerting operations;
- Immediate and first inerting renewal, then every 3 months inerting renewal of all containers;
- All data being written on a specific form (container reference, inerting renewal date, observations…).

For containers not coming from reactor block:

- Containers already equipped with specific caps allowing inerting operations;
- No immediate inerting operation, then every 3 months inerting renewal of all containers;
- All data being written on a specific form (container reference, inerting renewal date, observations…).

5.5. Tracking and labelling of the containers

Every sodium waste is identified and controlled before introduction in a container (all data being written on a specific form). The specific form to be filled for each sodium waste container allows:

- To characterize the objects:
  - (i) Reactor system (conventional or nuclear radiological aspect);
  - (ii) Mass of the objects;

- To characterize sodium:
  - (i) Reactor system (conventional or nuclear radiological aspect: tritiated and/or other radio elements);
  - (ii) Whatever sodium class: sodium film, bulk sodium, sodium residues;
  - (iii) Sodium film surface or sodium mass, in a container: total sodium mass and associated radioactivity (for safety analysis and storage place).

Each container is properly labelled. The associated form has to be filled during conditioning phase. Sodium waste storage area has to be marked out with signposts.

5.6. Transport

Sodium waste containers are transferred from conditioning area to storage room along a specific route. If this route is outside, its duration has to be shortened. In case of rain or snow (high rate of humidity), transfers are forbidden. Sodium waste containers can't remain outside.

5.7. Inerting procedure

The 213 L-drums and 10 L-stewpan can be inerted: they are equipped with a specific join in order to be connected with an argon bottle.

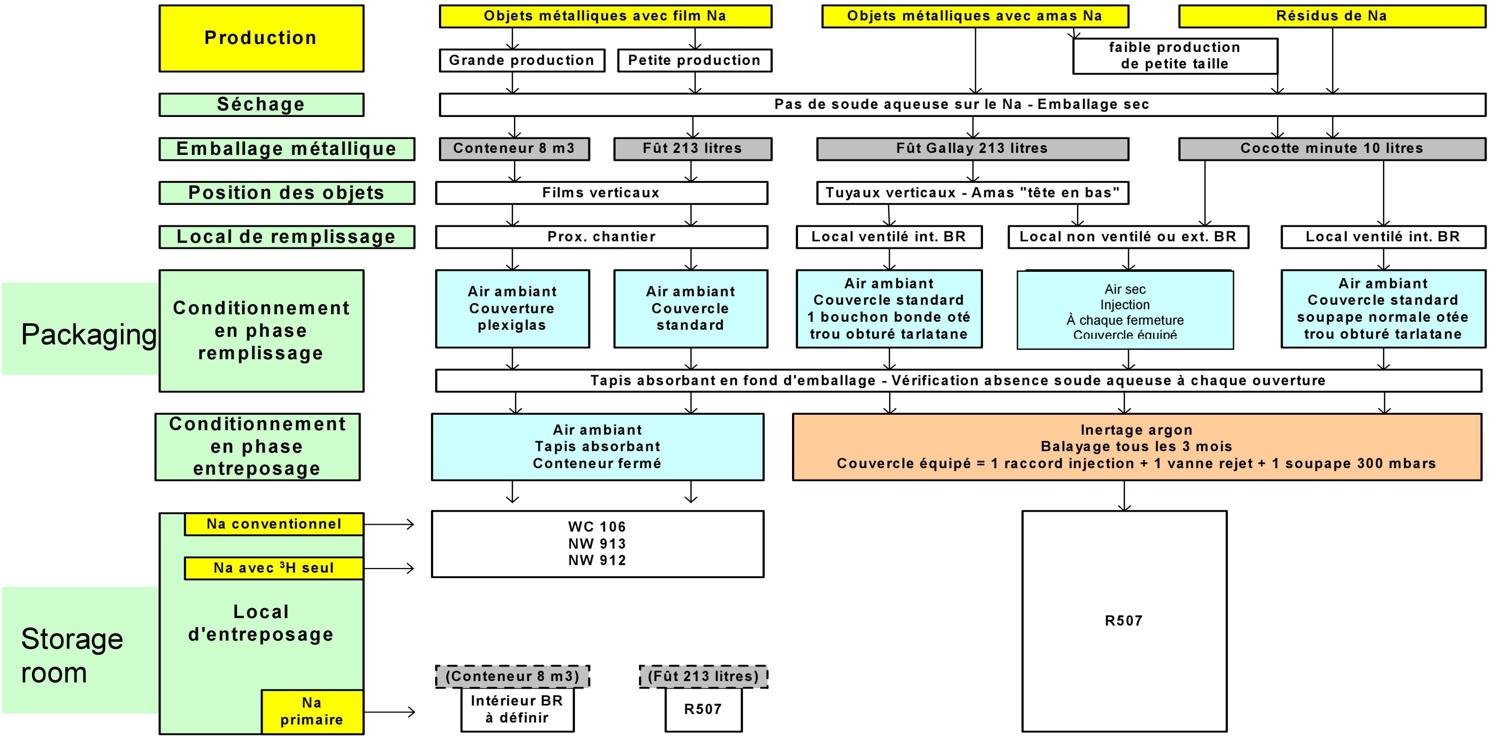

FIG. 9. General flowchart: Prescription s for conditioning and storage of sodium wastes and objects.

6. CONCLUSION

In conclusion, the kinetics of production of aqueous soda is very sensitive to humidity in air as well as temperature. For the cutting of small components containing only thin layer of metallic sodium, it was shown that secure operations (no production of aqueous soda and limited production of hydrogen) can be guaranteed over a period of 2 weeks if humidity is maintained below 3.7 g of water/kg of air, for a working temperature of 35°C. Anyway, production of aqueous soda should have limited effect regarding the limited amount of metallic sodium available. Maintaining the cut pieces vertically should help draining of aqueous soda and prevent non expected reaction.

For longer operations like cutting of large components containing bulk metallic sodium, it seems difficult to guarantee the absence of aqueous soda production (i.e. no reaction water/sodium and no production of hydrogen) unless working under dry air.

During these tests, it was shown that the amount of sodium available for reaction is not a criterion which guarantees the absence of aqueous soda production as there is a critical thickness of oxidized sodium (either Na_2O or NaOH). Below this critical limit (about 2-3 mm), diffusion of water and/or oxygen is possible and no aqueous soda is produced. But, above this limit, diffusion is too slow and aqueous soda is produced even if there is still metallic sodium available.

In conclusion, these tests pointed out that only a very low humidity is necessary to generate aqueous soda, which could react with remaining bulk metallic sodium. Hence, treatment of components under atmospheric conditions is safe only if humidity of air is removed or remains low enough to prevent the production of aqueous soda over the period of treatment. This kind of approach could be extended to the strategy of storage of cut components in trunks before washing if no inerting conditions are used. On this basis, new general safe rules for sodium waste storage have been raised:

- Waste classifying: pure sodium and soda to be separated, bulk and residues to be separated;
- Sodium waste containers: tight, dry, easy to refill with gas, protected against overpressure effect, with specific marking and reference;
- Dedicated rooms: dry, with specific markings, with specific sodium fire extinguishers;
- Maximum duration: three months before next refill with inert dry gas, in an over container if more than one year;
- Dry gas feeling: inert gas except for sodium film residues (dry air).

For Superphénix application, packaging and storage conditions of sodium wastes have been defined in detail, in accordance with container fluxes, from conditioning area to sodium waste treatment cell: it was decided to initially fill the containers (packaging phase) with dry air (dew point less than -10°C) whatever are the sodium waste types, because any gas in the container is rapidly ant totally dried by sodium itself. Then, renewal of gas in the containers will be done with dry argon (storage phase), except for sodium film residues (dry air filling). On site feedback experience will confirm the efficiency of these recent rules which have to be adapted to each specific case (sodium waste type, containers) and which can evolve with on site feedback experience.

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OVERVIEW OF SODIUM REMOVAL TECHNOLOGIES AND SELECTION OF REMOVAL METHOD FOR BN-350 RESIDUAL SODIUM DURING DECOMMISSIONING

S. ANDROPENKOV
NATIONAL ATOMIC COMPANY ‘KAZATOMPROM’, ALMATY, KAZAKHSTAN

Abstract

In this report the overview of the methods and the technologies, and the technical means and systems applied at the present time for removal the sodium residuals out of the equipment and pipelines is fulfilled. The comparative analysis of the different technologies from the points of view of safety, cost effectiveness and applicability has shown that for the conditions of BN-350 the most applicable is the relatively new, but very promising technology of passivation with a moist CO_2. In the present report it is shown the possibility of the use of the technology for the conditions of the BN-350 reactor. The main factor for the selection of the technology that is recommended for use at the BN-350 is the necessity of passivation relatively deep (about 10 cm) layers of the sodium. The positive results of the passivation of the residual sodium of EBR-II show the applicability of the hydrocarbonation technology for BN-350 reactor. The recommendations on the technologies of removal of the sodium residuals of the secondary and primary circuits of BN-350 are given too.

1. INTRODUCTION

One of the main tasks arising during fulfilling the activities on decommission of BN-350 fast breeder reactor is the task on removal the sodium residuals, left after completion of bulk coolant draining, out of the main reactor equipment. The necessity to solve this problem is defined not only by the radioactivity of the coolant, but by its high chemical (explosion and fire) activity too.

High chemical activity of the sodium determines the requirements on providing the special storage conditions of the drained equipment. In particular, it is necessary to keep continuously an inert gas media in order to prevent the possibility of reaction of the liquid metal residuals with the air humid. During reaction of the liquid metal coolant (LMC) with water the hydrogen gas and alkali are produced. The alkali reacts with the construction materials of the walls of the tanks and pipelines at elevated temperatures. This alkali corrosion may lead to equipment damage and following outflow of the LMC residuals to the environment (only if they are liquid). Apart of this, the collecting hydrogen may create the fire-dump gas inside of the drained vessels and than explode (only in the presence of oxygen).

Presently, in the world practice several technologies on residuals removal are developed. Using of the technologies depends on the particular start conditions. For BN-350 it is being required to develop and prove the applicability of a low risk technology of removal of the sodium residuals out of the loops and vessel of the reactor facility.

As the basis the gained practice experience and the results of the investigations on developing the technologies of ‘washing out’ of the alkali metal residuals out of the equipment and loops of other reactor facilities are taken, as follows:

- The condition of the reactor and loops of the primary and secondary circuits after draining;
- The short description of the existing methods of sodium residuals removal;
- The results of the comparative analysis of these methods;
- The recommendations of the technologies of residuals removal for the BN-350 conditions.

2. CONDITION OF THE REACTOR AND LOOPS OF THE PRIMARY AND SECONDARY CIRCUITS AFTER DRAINING

2.1. Reactor and primary circuit

Some amount of sodium remains as a film on inner surfaces of equipment and pipelines of sodium circuits after draining of sodium coolant. The existing experience of sodium equipment operation shows that ~ 0.1 L of sodium per 1 m^2 remains on inner surfaces of equipment in case of good sodium draining (dipping drift >10°). But there are some data that the thickness of the film is to be less than that (after draining EBR-II sodium, fastener threads and writing marks [made during fabrication] could be seen during inspection of the primary tank).

Also some amount of sodium remains as separated 'puddles' in non-draining pockets of equipment and pipelines. This volume is the sum of the following (Fig. 1):

- In the puddle on the bottom of the reactor vessel. The depth of the puddle is ~ 110 mm, so the volume is to be ~ 70 L (Fig. 2);
- In the puddle on the bottom of the central column. Undrainable sodium volume V is ~ 3 L (Fig. 3);
- In the cylindrical plenum A (Fig. 4). Undrainable sodium volume V is ~ 20 L;
- In area B in the high pressure plenum (Fig. 4). Undrainable volume V≈ 10 L;
- A problem is the possibility of the sodium flow, during the vessel drainage, through the cylindrical gap between the cone wall of the cooling tract and the flange of the high pressure plenum. Dependently of the actual value and the condition of the gap ($\delta_{max} \approx 3$ mm and $\delta_{min} \approx 0.5$ mm), the cylindrical volume may not be drained (undrainable volume V ≈ 250 L);
- ≈ 60 L of film residuals of 30 μm on practically vertical wet surfaces of inner reactor structures.

Having ~ 6 000 m^2 of inner surface of primary circuit including fuel subassemblies mock-ups, on surfaces of equipment and pipelines of the primary circuit will remain ~ 600 L as the film residuals (including ~ 60 L of film residuals on practically vertical surfaces of inner reactor structures) and ~ 353 L as the locally located puddles of sodium inside of the reactor vessel.

All equipment of loops of the primary circuit (MCP, LST, non-return valves) is drained practically to the full, except main isolation gates D_n-500, D_n-600. There is undrainable pocket in the each vessel of the gate (Fig. 5). Undrainable sodium volume V is ~ 7 L in the each gate. Altogether in the 12 gates of the primary circuit after draining are left ~ 80 L of sodium.

Each of five cold traps of the primary circuit will contain after draining ~ 3.2 m^3 of sodium with accumulated impurities including radioactive ones. In each of 10 drainage tanks of the primary circuit on the inner surfaces and bottoms will remain ~ 20 L of sodium. Altogether in the primary circuit will be left after draining ~ 22 m^3 of sodium. It is necessary to mention that before the passivation of the primary sodium residuals all cold traps and drainage tanks must be cut off (cut off and welded up) from the circuit. The technology of removal the sodium residuals out of them is a subject of a separated project. The amount of the sodium residuals to be passivated in the primary circuit of the BN-350 reactor facility is about 0.6 m^3 in the films on the loops walls, and 0.433m^3 as the puddles and pockets.

2.1. Secondary circuit

In each of six loops of the secondary circuit after draining will be left ~ 0.65 m^3 of sodium on inner surfaces of equipment and pipelines. In non-draining matrices of intermediate heat exchangers of every loop of the second circuit will remain ~ 6.7 m^3 of sodium. In each of six cold traps of the second circuit will be left ~ 3.2 m^3 of sodium and in each of drain tanks of the second circuit will be left ~ 20 L of sodium. Altogether in the second circuit will be left after draining ~ 63 m^3 of sodium.

Before the passivation of the primary and secondary circuits, all cold traps and drainage tanks must be cut off (cut off and welded up). The sodium removal technology for these volumes is an issue of separate project and is not described in the present report. Removing the residuals out of the IHEs is a separate task too, and it is not described in the present report. The amount of the secondary sodium to be passivated is ~ 0.65 m^3. The loops are to contain ≈ 0.2 m^3 of the film.

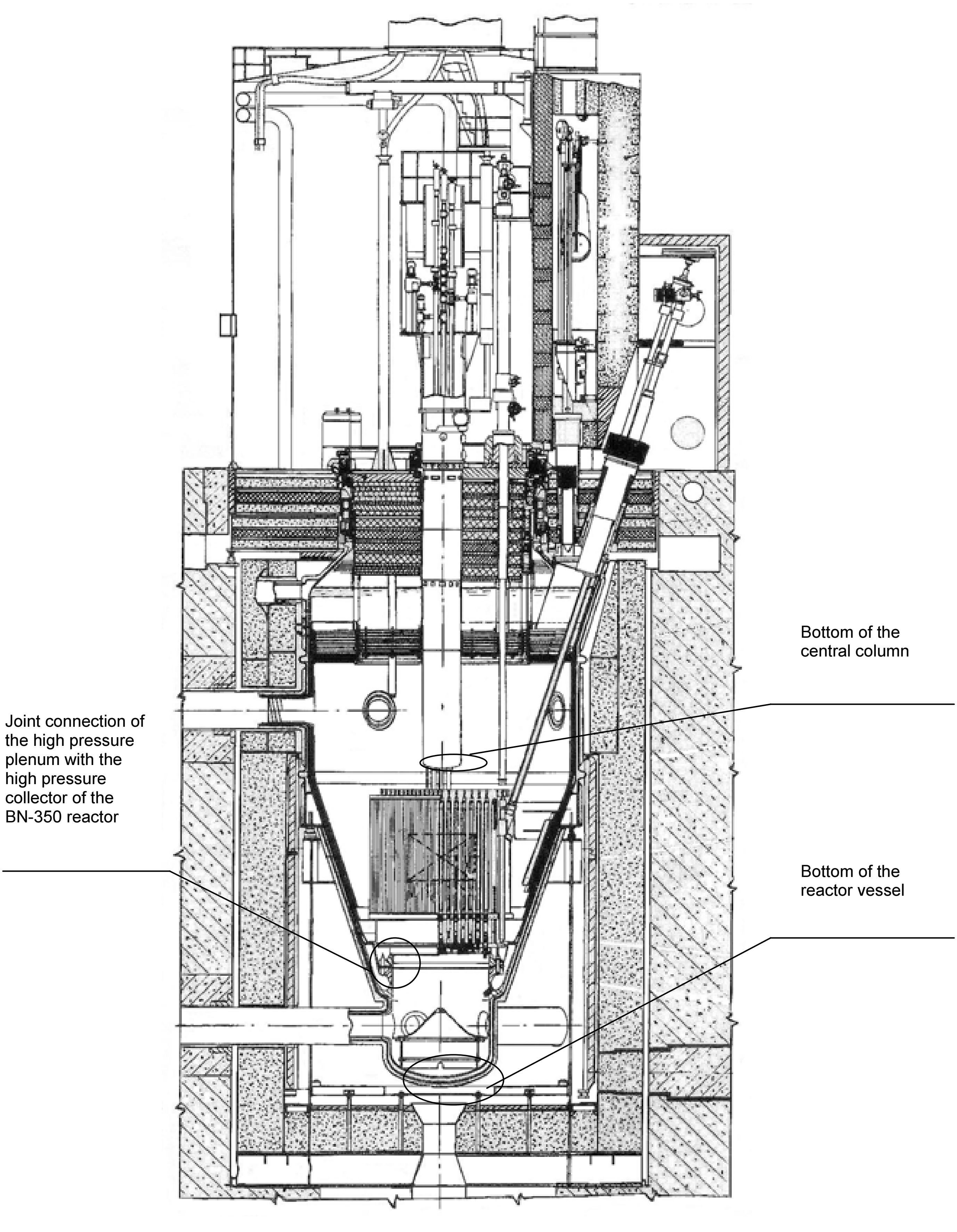

FIG. 1. Areas of locally located residual masses of the coolant in the BN-350 reactor after draining.

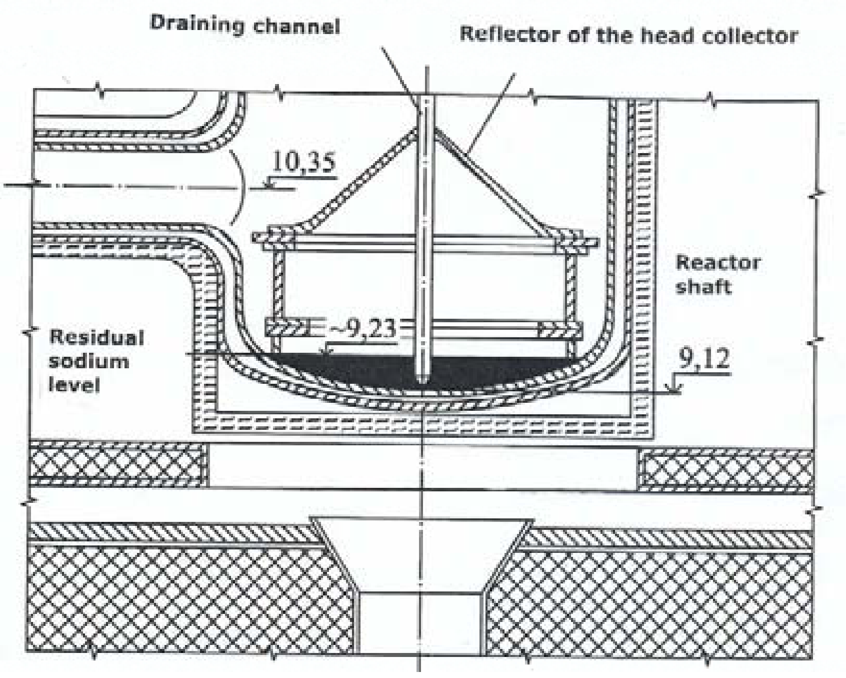

FIG. 2. Residual sodium on the bottom of the reactor.

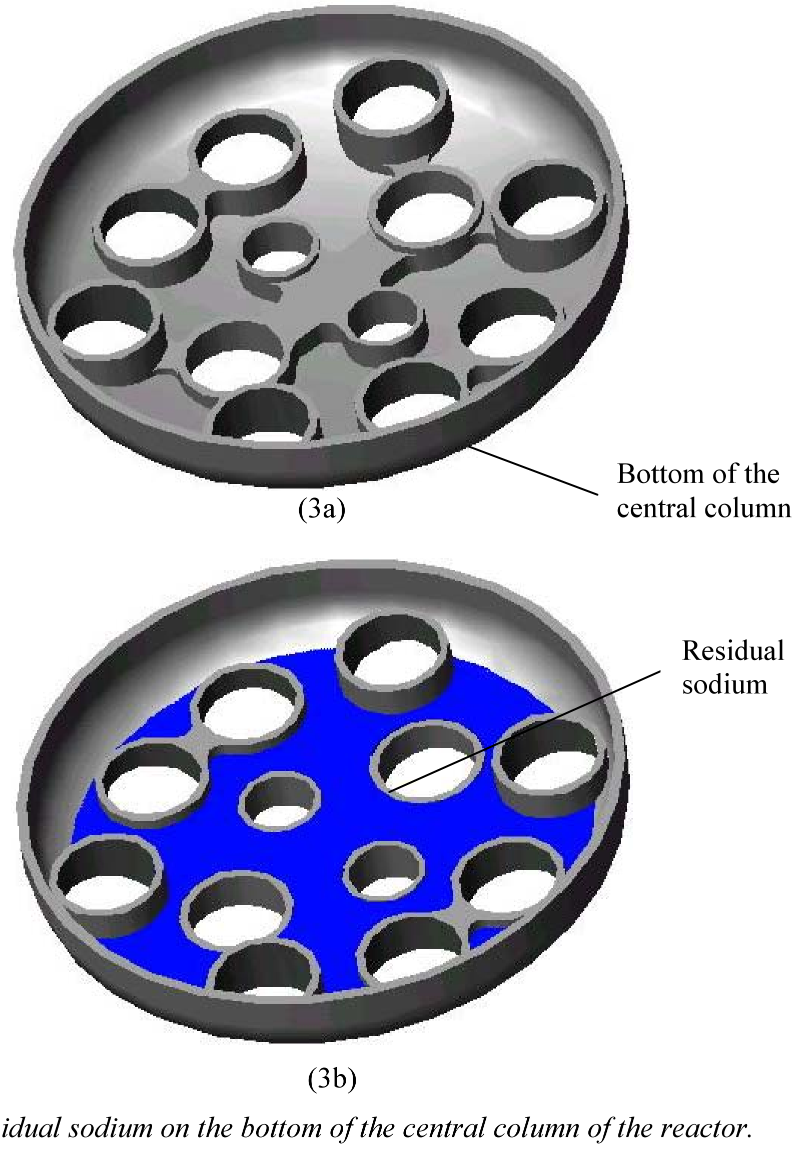

FIG. 3. a,b. Residual sodium on the bottom of the central column of the reactor.

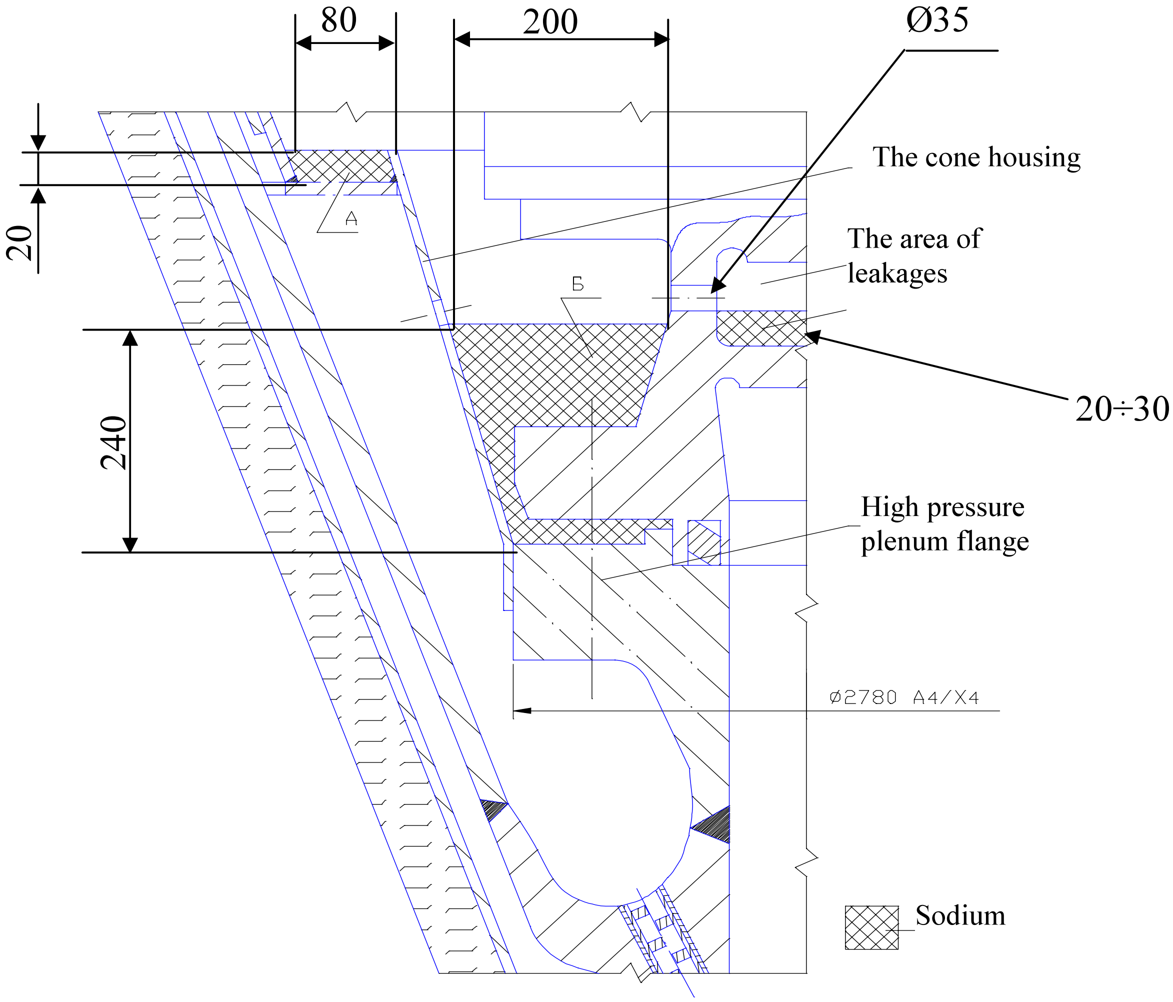

FIG. 4. The point on connection of the high pressure plenum with the high pressure collector of the BN-350 reactor.

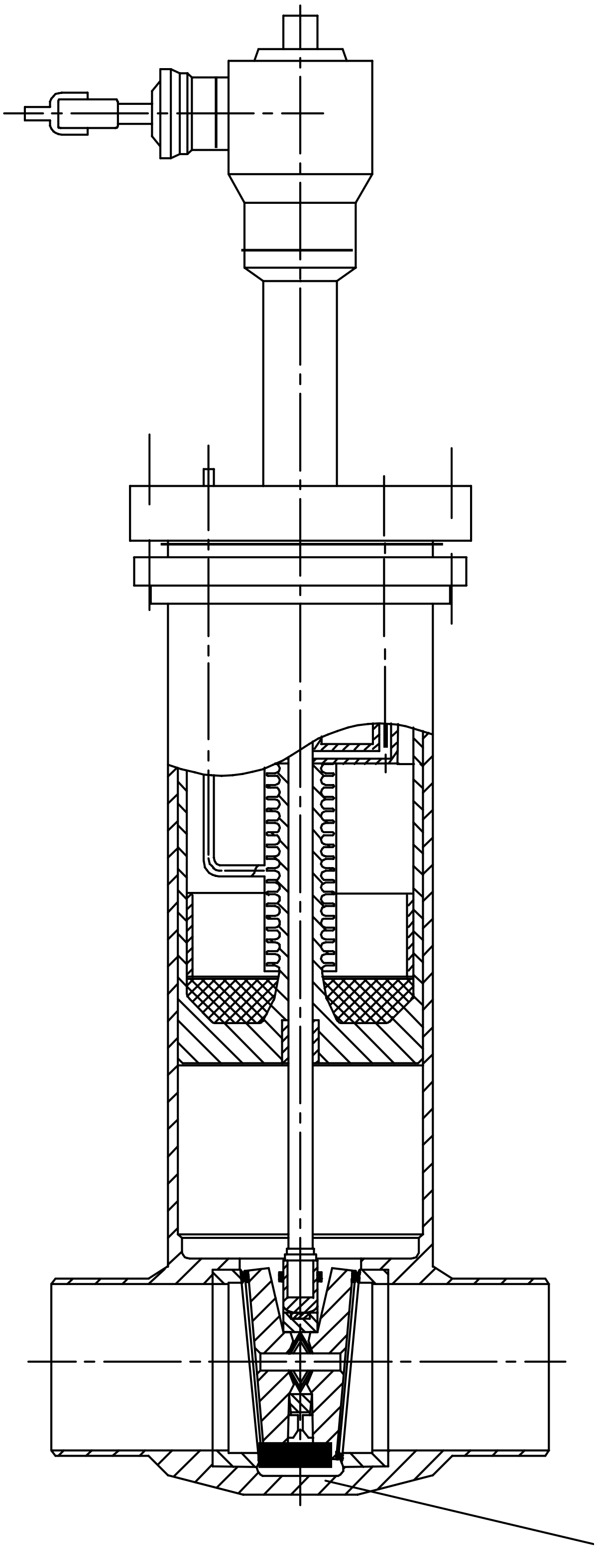

FIG. 5. Residual sodium in undrainable pocket of the gates D_n-500, D_n-600.

3. DESCRIPTION OF THE MAIN TECHNOLOGICAL PROCESSES, TECHNICAL MEANS AND SYSTEMS USED FOR SODIUM RESIDUALS REMOVAL AND RECOMMENDATIONS ON THE PRACTICAL APPLICATION OF THE OPTIMAL METHOD FOR THE BN-350 REACTOR CONDITIONS

A number of methods of alkali metal residuals removal out of the equipment and systems of reactor facilities are used at the present time. In this report the following methods and technologies are taken in consideration:

- Washing with water under vacuum;
- Washing with water alkali solutions under vacuum;
- Washing with dispersed water (water fog);
- Steam-gas washing (vapor-nitrogen washing);
- Water-oil washing;
- Washing with light alcohols;
- Washing with heavy alcohols;
- Vacuum distillation;
- Hydrocarbonation technology (water saturated/moist CO_2);
- Washing with melted alkali solutions.

Requirements raised by practice of experimental sodium stands operation leaded to development of steam-gas method of equipment and circuits cleaning both experimental stands and functioning BN type reactors. Experience showed sufficient effectiveness, safety and economy of this method.

However, it was found that this method does not cover the whole spectrum of qualifying standards of the washing technology of different kind of equipment, design, operation conditions etc. The necessary of solving these tasks leaded to development of alternative methods of cleaning aimed to exclude the following drawbacks of the steam-gas method:

- Possibility of corrosive damage;
- Possibility of mechanical damage;
- Possibility of blocking of thick sodium layer by sodium hydroxide resulting from interaction with steam. During following washing with water the layer of sodium hydroxide is dissolved and sodium reacts with excess water, sometimes violently; hydrogen exits intensively that in case of contact with air can lead to creation of hydrogen-air mixture of explosive concentrations.

Corrosive damage of structure materials is practically excluded when methods of washing with water mist, water-vacuum, alcohol (light and heavy) and vacuum distillation are used. Washing with alcohol is extended conformably to equipment and devices having low mechanical strength.

Exclusion of possibility of forming of blocking layer of sodium hydroxide appeared on sodium surface during water-gas washing is possible when methods of alcohol and water-vacuum are used because sodium hydroxide is well dissolved in water and alcohol. Finally, the vacuum distillation method has doubtless advantage: there is no hydrogen release. At the same time, each of the methods has its own drawbacks, demands maintenance of specific conditions and determined range of possible application.

The corrosive damage of the structural materials of BN-350 is not the most important factor for estimation of the applicability of the technologies of removal of the residual sodium. Since, there is not intention to use the main technological equipment of BN-350 after the residuals removing.

Presently there are several projects on decommission of large reactors using the alkali metals as the coolant; and the works on developing the new technologies of safe and effective LMC residuals removal out of the vessels and pipe-lines of the large sized reactor are being fulfilled.

The large volumes of the washed equipment (hundreds m^2) with complex inner surface (thousands m^2) and large amount of the LMC residuals (tons) bring the additional requirement to the technologies and methods. The work on removal or passivation 'in situ' the BN-350 residuals are a work with a large-size reactor. So, the technologies of 'in situ' passivation or removal the LMC residuals that have been developed for small-size equipment must be re-estimated with considering the new scales of the application for the BN-350 conditions.

The comparative analysis of the abovementioned technologies from the points of view of safety, engineering feasibility, cost effectiveness and minimizing of derived radioactive waste amount has shown that for the conditions of BN-350 the most applicable are:

— The technology of the steam-gas washing;
— The new, but very promising technology of passivation with a moist CO_2.

Main stipulation of the possibility to use the steam-gas washing technology is the sodium draining of equipment and pipelines to the full or the cutting off selected volume from the areas with the residual sodium.

At present time there is no experience on the using of the steam-gas washing technology for the equipment and pipelines compared on the scale and complexity with BN-350 reactor. In addition, as it was shown in chapter 1, there is no practically possibility to drain sodium from the equipment and pipelines to the full.

Taking into account the significant temperature and speeds of chemical reactions in process of the steam-gas washing, possibility of alkali cracking of the construction materials, failures of the integrity of the circuits and generation the hydrogen and oxygen fire-dump mixture, the safety level of the steam-gas washing technology for BN-350 reactor conditions could not be considered as satisfactory. Additionally the significant amount of derived radioactive wastes is produced in the process of the steam-gas washing. These radioactive wastes should be treated.

Another approach to the methods of sodium residuals removal for LMC type reactors is the relatively new one named 'passivation' (or another name 'hydrocarbonation'). The idea of this method consist of creation of the circulation of the CO_2 gas mixture with little amount of H_2O vapor inside the treated volume. During fulfilling the process the dew – point should be controlled. Sodium slightly reacts with H_2O with producing anhydrous sodium hydroxide. That anhydrous sodium hydroxide in its turn reacts with CO_2 producing sodium carbonate Na_2CO_3. Accordingly the next main chemical reactions are carried in the process of passivation:

$2\ Na + 2\ H_2O = 2\ NaOH + H_2$,
$2\ NaOH + CO_2 = Na_2CO_3 + H_2O$,

As the result chemically indifferent Na_2CO_3 is produced. Adjusting the gas mixture humidity the concentration of the hydrogen produced in the process of the reaction could be sustained on the safety level: from the tenth of percent up to few (2%) percents.

The main advantage of the method is exception of producing liquid sodium hydroxide. As in case of the other technologies of this type the liquid sodium hydroxide as a film prevents in certain time the moisture access to the unreacted sodium, but then the film of the liquid sodium hydroxide is dissolved and the uncontrolled reaction of the sodium with water is beginning.

One more advantage of the method, relating to the BN-350 reactor conditions, is no necessity to dispose the final products of the hydrocarbonation. Since Na_2CO_3 is chemically indifferent and practically harmless in SAFESTORE conditions of the BN-350 reactor (conditions of no interaction between the reacted sodium and atmosphere air).

The main question of usage only the passivation technology is the possibility of reacting of the residual sodium to the full. Only factor restricting the efficiency of the passivation method is the reducing of the chemical reaction speed due to increasing of the layer thickness of the interaction products that prevents access of the working reagent (the CO_2 gas mixture with little amount of H_2O vapor) to the unreacted sodium.

For solution of the question the provisions of the passivation process intensification and the substantiation of explosion and fire safety in case of conditions of no interaction between the reacted sodium and atmosphere air when the residual sodium was reacted not to the full were developed.

The recommendations on the practical application of the passivation method for the BN-350 reactor conditions are given below.

3.1. General provisions

As it was above mentioned the passivation reaction speed of the residual sodium strongly depends on the hydrocarbonate layer thickness. If provided with the on-stream inflow of the chemical reagents to the interface boundary of the sodium – hydrocarbonate with different speeds of the chemical reagents flow then the maximum passivated sodium deepness could be significantly increased and the time of the passivation process could be decreased. The on-stream inflow of the chemical reagents to the surfaces of the locally located residual masses of the sodium would be implemented by means of the special channels. The intensification of the passivation process after formation of the interaction products 'coat' would be performed using the different speeds of the chemical reagents flow to the surfaces.

The passivation process of the residual sodium in the reactor vessel and the pipelines of the primary circuit consist of the two main steps:

- Selected (and simultaneously, if it would be technically feasibly) passivation of the relatively large puddles in the reactor vessel and film residuals and small-sized locations of the residual sodium on the walls of pipelines of the primary circuit;
- Combined passivation of the residual sodium in the reactor vessel and the pipelines of the primary circuit.

The dividing of the passivation process on two steps, of course, is relative. In point of fact this method presents the series-parallel process for BN-350 reactor conditions.

The first step of the passivation process would be performed separately in the reactor vessel and in the each loop of the primary circuit. For that end all main isolation gates of each loop should be closed. Of course there is not full leaktightness of the main isolation gates due to presence of some solid sodium droplets on the surfaces of the gates seats. But there is not necessity to provide the full leaktightness because the first step would be performed in 'steady-state' mode (without organization of the reagent circulation through the reactor vessel and the loops of the circuit). The relative leaktightness is necessary only for controlling the amounts of the reacted sodium by means of control of the amounts of hydrogen in exhausted channels from each loop of the primary circuit and from the locally located residual masses of the sodium in the reactor vessel.

During the second step of the passivation process the main isolation gates are opened and the induced gas circulation through the loops and the reactor vessel is performed by means of turning on the MCPs in the mode of the fan. The different speeds of reagent flow would be variated by means of changing quantity of the working MCPs, speeds of its revolution, and extent of opening/closing of the main isolation gates.

3.2. Passivation of the locally located residual masses of the sodium in the reactor

The passivation of the locally located residual masses of the sodium in the reactor would be implemented by means of the special channel installed in the auxiliary penetration of the refueling mechanism. Since the auxiliary penetration of the refueling mechanism is situated on small rotating plug the universal channel with changeable spray heads could be directed to any location of the residual sodium by means of rotation of small and big rotating plugs. For passivation of the residuals on the bottom of the central column the penetrations for mechanisms of the control and protection system would be used for installation the inflow and exhausted channels (Fig. 6). Using of the universal channel with changeable spray heads permits close the head to the surface of the residuals as minimum as possible distance without drilling the reactor internals (Figs 7 and 8). In case of necessity the penetration for a sodium level gage could be used for installation of the channel (Fig. 9).

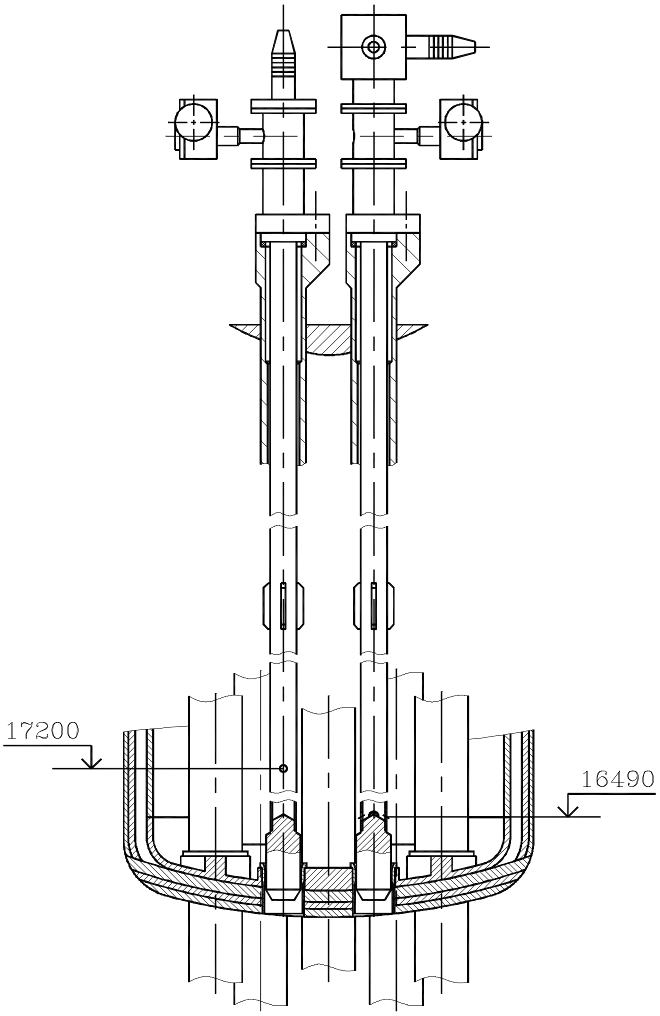

FIG. 6. Inflow and exhaust channels for passivation of the sodium residuals on the reactor central column bottom.

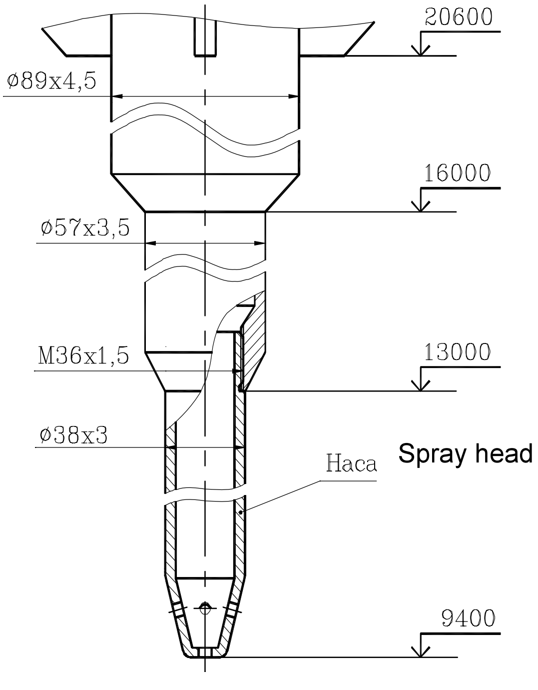

FIG. 7. Channel spray head for reagent inflow to the sodium residuals on the reactor vessel bottom.

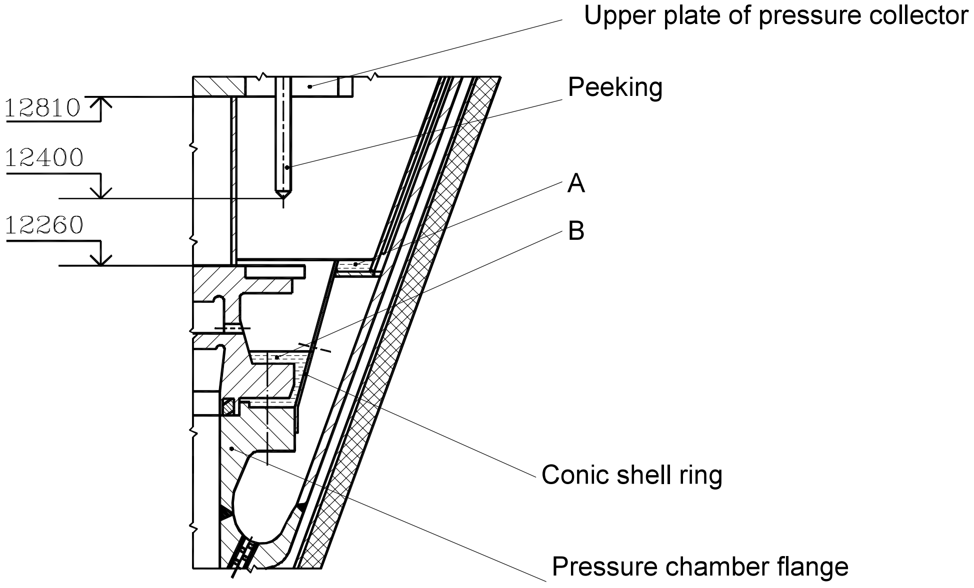

FIG. 8. *Inflow channel for passivation of the sodium residuals in the area of connection of the high pressure plenum with the high pressure collector of the BN-350 reactor.*

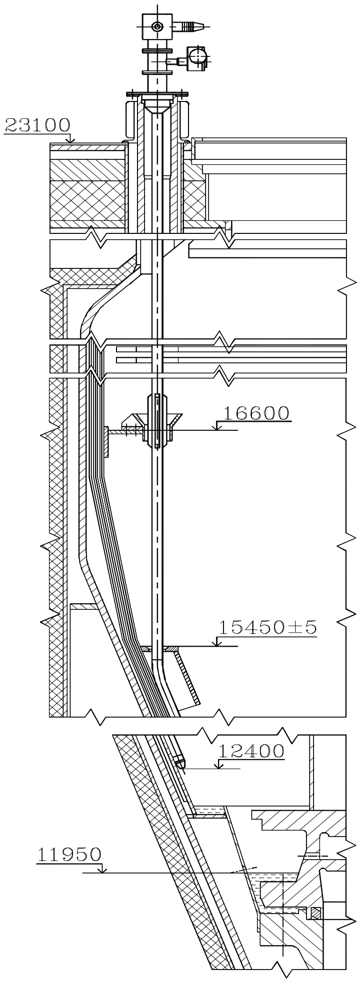

FIG. 9. *Inflow channel for passivation of the sodium residuals in the area of connection of the high pressure plenum with the high pressure collector of the BN-350 reactor installed instead of the sodium level gage.*

SECONDARY COLD TRAPS TREATMENT AT SUPERPHENIX

O. CARRA, R. BLACHIER, G. DEPREZ
FRAMATOME ANP, LYON, FRANCE

E. JOULIA
ELECTRICITE DE FRANCE (EDF) – CIDEN, VILLEURBANNE, FRANCE

Abstract

Secondary cold traps treatment is part of the general cleanup and dismantling of the fast breeder reactor Superphénix (Creys-Malville – France) which was shut down in 1997. The purpose of the treatment is to remove sodium hydride and sodium oxide from the inside wire mesh which was used as a filter. Neutralization is accomplished by using slow and gradual chemical process. Fifteen cold traps must be removed from secondary loops of the reactor. These components can be sorted in two categories: six small cold traps which contain approximately 100 g of sodium hydride, 5 kg of sodium oxide and 3 kg of residual sodium; and nine large cold traps which contain approximately from 15 to 200 kg of sodium hydride, from 80 to 300 kg of sodium oxide and 70 kg of residual sodium. Four small cold traps and four large cold traps are contaminated in tritium, some of them highly. A cold trap treatment process has been studied and is under development. It can be divided in four phases:
Hot treatment phase (thermolysis): vessel is maintained at high temperature while a constant flow rate of dry hot nitrogen passes through the cold trap. During this phase, sodium hydride is dissociated in hydrogen (tritiated, if the cold trap contains tritium) and sodium. Hot water vapour nitrogen: cold trap is heated while a constant flow rate of wet nitrogen is maintained in the cold trap, in order to neutralize sodium oxide, sodium and possible residual sodium hydride. After cooling, caustic soda is dissolved by water, during different phases. The treatment is finished by drying cold trap vessel using dry nitrogen. The aim of the process is to achieve all the treatment inside the cold trap vessel, without creating any other waste. Hydrogen rate in process gas exhaust will be maintained at low level during all the phases. A gradual experimental validation program of this process has been established. Controlled dissociation of sodium hydride using thermolysis will be first started with a small quantity of product, using an experimental vessel. The six small cold traps from SPX will be used for a pilot scheme of the complete process. An experimental device will be built in order to perform trials. At the end of each trial of a new process phase, analysis and inspections will be performed. The first treated cold trap will be dismantled at the end of cooling phase in order to perform analysis of the final products. Then, a final device will be built to treat secondary large cold traps.

1. COLD TRAPS DISMANTLING PROBLEM

In sodium FBR in operation, the function of cold traps is to purify the sodium. Thus, the cold trap has to remove:

- Oxygen, mainly resulting from pollutions following handling or incident (air entry, …). So oxygen is mainly found in the primary circuit and is caught in the cold trap as sodium oxide Na_2O (solid);
- Hydrogen, for instance found at a certain level in the secondary loops, due to diffusion of hydrogen through pipes of SG (especially thermal decomposition of hydrazine N_2H_4 from water of the tertiary loop). Hydrogen is caught in the cold trap as sodium hydride NaH (solid).

After sodium draining, cold traps enclose sodium hydride, sodium oxide and remaining sodium, all products that are chemically very reactive (they react vigorously with water, oxygen, halogen, alcohol…). Therefore, the way to proceed is to neutralize those products before mechanical dismantling of the traps. More precisely, aims of a cold trap process for its dismantling could be summarized as follows:

- Elimination of chemically reactive products (NaH, Na_2O, Na), producing effluents (liquid, gas) acceptable and, in the end, releasable;

— Prevention of any hazard, especially a violent chemical reaction (sudden reaction of a quite great amount of NaH and/or Na_2O and/or Na, hydrogen risk,…) or a contamination (leak, excessive release…).

A process for a cold trap dismantling is less difficult to establish than a process for its regeneration: in this latter case, the trap integrity must be, in addition, saved, especially its mechanical integrity; for instance, corrosion must be limited as far as possible.

1.1. SPX cold traps to process

In the frame of SPX dismantling, different cold traps are to be processed, some are middle size cold traps, and the others are huge size cold traps. The cold traps have quite the same configuration (setting aside their sizes), see Fig. 1. They include an economizer in upper part, a wire mesh cartridge in lower part and an outside cooling circuit on their side (organic liquid loop plunged in a liquid thermal conductor -NaK- in the case of huge size traps, air in the case of middle size traps).

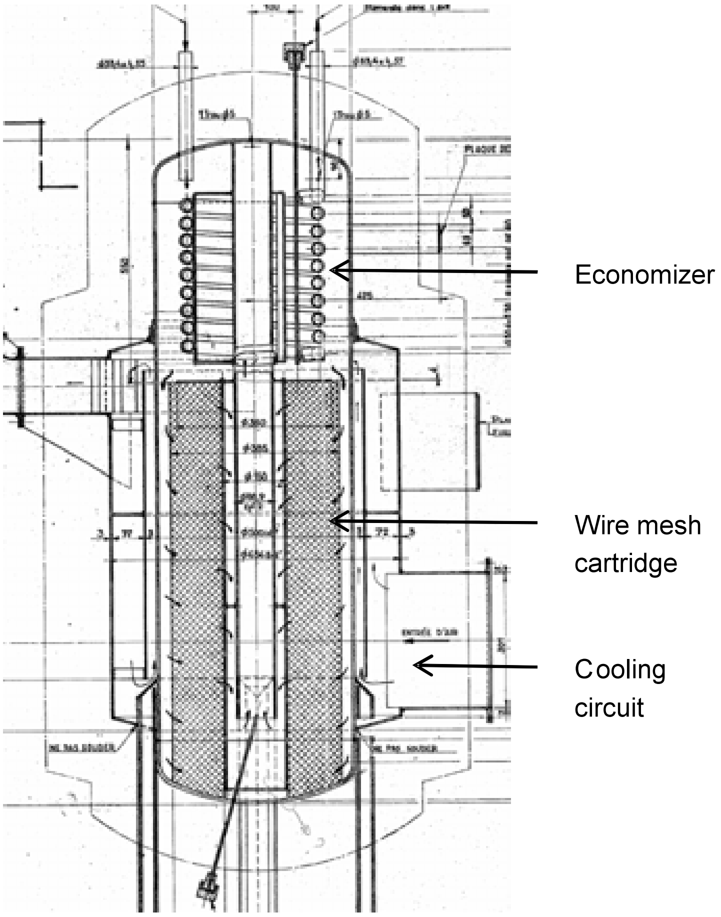

FIG. 1. SPX cold traps configuration.

Cold traps operate in the following manner: the sodium is cooled inside the trap; the solubility of hydrogen and oxygen in sodium decreases then and the hydride and oxide precipitate in the wire mesh cartridge that offers a wide area per volume unit (around 350 m^2/m^3).

1.2. Principles of the planned process

Essential phases of the planned process are performed at ‘high’ temperature (at a temperature higher than the ~ 320°C fusion temperature of sodium hydroxide -and a fortiori the ~ 98°C fusion temperature of sodium).

Furthermore, process parameters are set so that the rate of hydrogen in the exhaust gas of the processed trap is maintained under the flammable low limit of hydrogen in air.

1.2.1. First phase: thermolysis

This phase occurs during the heating of the trap and then during its maintenance at high temperature by the mean of heat tracing cables (and thermal insulation). A warm nitrogen sweeping flow is at first established through the trap.

During this phase, sodium hydride is dissociated according to the reaction $NaH (s) \rightarrow Na (\ell) + \frac{1}{2} H_2 (g)$.

Liquid sodium remains in the trap and the hydrogen is released in the gas stream at the exit of the trap. The tuning of the processing temperature and of the sweeping flow rate allows to control the hydrogen emission, in a way that, as already said, the rate of hydrogen in the exhaust gas is maintained under the flammable low limit of hydrogen in air. The exhaust gas is thus never flammable, and in particular when it is diluted in air. This arrangement allows mastering the hydrogen risk.

The $NaH \rightarrow Na + \frac{1}{2} H_2$ reaction is endothermic at any temperature and thus it cannot diverge. But, its kinetic strongly rises with the temperature T and reactive surface (interface between Na-NaH and gas).

A hydrogen equilibrium pressure settles in a gaseous closed sky covering some Na-NaH [1]. When this equilibrium pressure is reached, no more sodium hydride dissociates. At a given temperature, if the sky is swept with an inert gas, it is sure that the hydrogen partial pressure in this sky is lower than the equilibrium pressure at this temperature and so the reaction is going on. Taking into account the low hydrogen partial pressure in the gas stream, the reaction $Na_2O + \frac{1}{2} H_2 \rightarrow Na + NaOH$ will not occur, contrary to what happened in some past cold traps experiments [2]. The end of phase criteria will more or less be:

- No more hydrogen emission, or very small;
- Thermolysis will have lasted sufficiently, regarding the assessed amount of sodium hydride in the trap at the beginning of the processing and the nitrogen flow rate and the hydrogen rate in the exhaust gas.

In case of unexpected situations, the thermolysis is interrupted switching off the heater of the processing gas and the heater of the trap. In the past, several thermolysis operations of cold trap have been performed [2]. The processing temperatures were far higher than the one planned here, as well as the rates of hydrogen in the gaseous effluent. It is a challenge to carry out a cold trap thermolysis while keeping the hydrogen level low in the exhaust gas.

1.2.2. Second phase: hot WVN

This phase consists in the continuity of the previous, to moisten, in an accurate manner, the nitrogen sweeping of the trap (injection in this nitrogen of vapour water), so that the following total reactions of neutralization of sodium and derivatives with water arise:

$Na_2O (s) + H_2O (g) \rightarrow 2NaOH (L)$,

$Na (L) + H_2O (g) \rightarrow NaOH (L) + \frac{1}{2} H_2 (g)$,

$NaH (s) + H_2O (g) \rightarrow NaOH (L) + H_2 (g)$.

Liquid sodium hydroxide remains in the trap and the hydrogen is released in the gas stream at the exit of the trap. The rate of water vapour in the processing gas is limited, so that, even if all the water vapour reacts, the hydrogen rate in the exhaust gas is under the flammable low limit of hydrogen in air. This, while the carrying gas is, as in the thermolysis, nitrogen and not air. This arrangement, if fulfilled, cancels hydrogen risk. Conversely, the drawback of the choice of low rate of hydrogen in the gaseous effluent is to lead to a long duration of the processing (the processing is slow).

At a temperature higher than its fusion temperature, sodium hydroxide hydrates very few, even in a very wet atmosphere (it thus stays practically anhydrous). Therefore, the risk of violent reaction between aqueous caustic soda and bulk Na-Na_2O-NaH, which exists in WVN at lower temperatures, is withdrawn. Likewise, given the processing temperature is very superior to 100°C, there is no possibility of condensation of water of the processing gas, liable to lead to a violent reaction with a sodium and/or derivates pool. Due in particular to the low hydrogen partial pressure, sodium-sodium hydroxide reaction could occur in the hot WVN phase, such as:

$Na (L) + NaOH (L) \rightarrow Na_2O (s) + \frac{1}{2} H_2 (g)$,

$2 Na (L) + NaOH (L) \rightarrow Na_2O (s) + NaH (s)$.

In fact, those reactions are chemical equilibria. For instance, the inversion temperature θ of the second reaction is around $\sim 420°C = \theta$, with Gibbs's function $\Delta G^0(T) < 0$ if $T < \theta$ and $\Delta G^0(T) > 0$ if $T > \theta$. This second reaction is used to eliminate sodium hydroxide in sodium during temperature increase of a sodium cooled FBR at its start-up or restart up: a thermal plateau is made between 320°C and 420°C, with sodium purification operating, so that the possible caustic soda inside the sodium could react with this sodium, producing hydride and oxide, which are then trapped by the cold traps. Those reactions should be of limited impact: expected kinetics is slow, in accordance with some papers of the literature.

With the chosen parameters of the planned process, there is no risk of Caustic Stress Cracking of the austenitic stainless steel of the cold traps, at least with respect to the classical domain of this corrosion of this steel with aqueous caustic soda (see Fig. 2; a priori, temperature will be too high and the caustic soda will be too dry). Nevertheless, a quite comprehensive bibliographical study of caustic corrosion in the processing conditions (austenitic steel with possible stress -welds-, elevated temperature, concentrated sodium hydroxide, inert atmosphere with water vapour) was carried out; this study does not exhibit manifest evidences of harsh caustic corrosion in those conditions. However, preliminary tests will be performed to investigate this specific point.

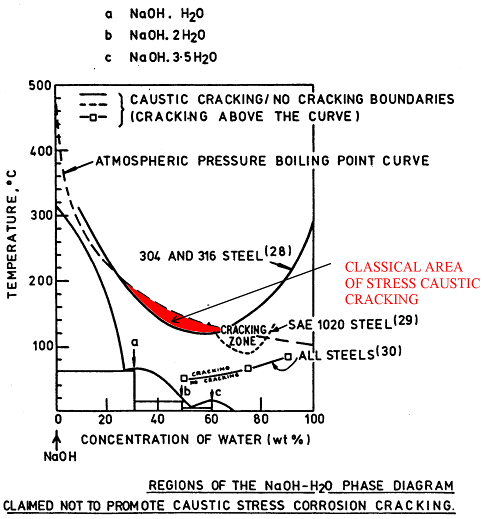

FIG. 2. 'Hoffman' diagram.

The end of phase criteria will be more or less:

- No more hydrogen emission, or very small;
- No more hot spot at the places of pools of sodium and derivatives (for instance vessel bottom head);
- Hot WVN will have lasted sufficiently, in regard to the assessed amount of sodium, of sodium oxide and of remaining sodium hydride in the trap at the beginning of this hot WVN, and also with respect to the sweeping gas flow rate, and the chosen rate of H_2O in this sweeping gas and the expected efficiency of the processing, in term of consumption of the injected H_2O.

In case of unexpected situations, the hot WVN is interrupted switching off the injection of water vapour. The WVN process for sodium wetted components or vessel is well-known, but it is generally performed at a lower temperature. On the following diagram 3 of the caustic soda phases (abscissa: temperature, ordinate: water vapour partial pressure), WVN at low, middle and high temperature have been positioned.

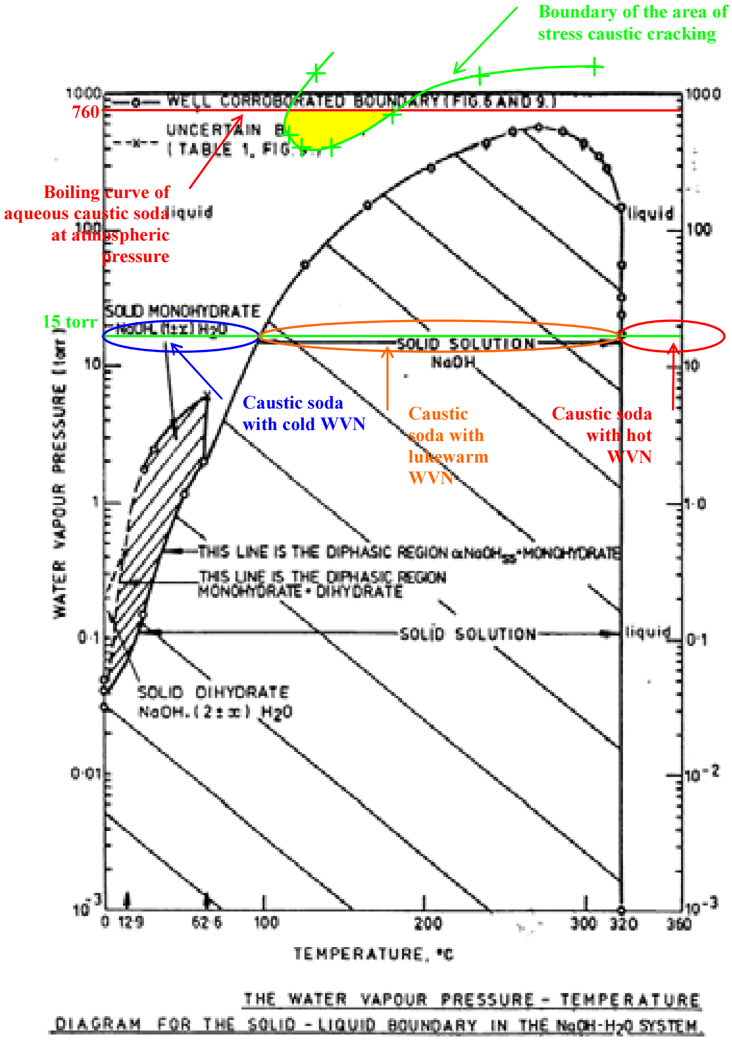

FIG. 3 Diagram of the caustic soda phases.

High temperature for WVN means at a temperature higher than the fusion temperature of caustic soda (~ 320°C). In this case ('hot WVN'), and as seen above, the generated caustic soda is liquid and anhydrous.

Middle temperature for WVN means at a temperature lower than the fusion temperature of caustic soda (~ 320°C), but enough high so that, in this case ('lukewarm WVN'), the generated caustic soda is solid and anhydrous. The low value of the temperature range for lukewarm WVN depends of the rate of water vapour in the processing gas, the higher this rate is and the higher this low temperature value is. In practice, for low water vapour rate in the processing gas (a few % V), the low value of the range for lukewarm WVN is more than 100°C.

Low temperature for WVN means at a temperature enough low so that, in this case ('cold WVN'), the generated caustic soda is, at equilibrium, aqueous (and consequently liquid). Again, the high value of the temperature range for cold WVN depends of the rate of water vapour in the processing gas, the lower this rate is and the lower this high temperature value is. In practice, for low water vapour rate in the processing gas (a few % V), the high value of the range for cold WVN is less than 100°C. So WVN at ambient temperature is cold WVN. Hot WVN, and prior thermolysis, generate only fluid products (sodium in thermolysis, liquid caustic soda in WVN, and hydrogen in both).

First, this avoids the plugging of the processed component (the cold trap in the case considered in this paper) that could occur with a process that generates solid product (as for instance lukewarm WVN or carbonation). Second, with hot WVN, the full reduction of pools is promoted.

The produced caustic soda flows to the bottom of the trap, or, if it stays in the area of the pool where it was generated and if it does not react with the sodium, it sinks under this one, as caustic soda is more dense, or the caustic soda and the sodium solves one in the other, allowing in any case that the contact between sodium and water vapour of the processing gas remains.

There is no formation of a solid layer that could potentially prevent the water vapour of the processing gas from reaching products to be water neutralized, as in some WVN processes at lower temperature or in carbonation process that could give or give rise to the formation, besides the gaseous hydrogen, of a solid product.

As seen above, with lukewarm WVN, solid caustic soda is generated on principle. So this process has the drawback above mentioned. This leads to very slow and irregular sodium destruction with it. Conversely, there is absolutely no risk of caustic corrosion with this process, as the generated caustic soda is solid.

With cold WVN, aqueous caustic soda is generated in principle, but, due to the fact that destruction of sodium is exothermic, this generated aqueous caustic soda could locally and temporarily dry, making an insulating crust.

With carbonation, sodium carbonates, which are solid products (with however a certain porosity), are generated. The interest of carbonation, in comparison with cold WVN, is, as caustic soda is converted into carbonates, to get rid of the risk of a flow of aqueous caustic soda, which may cause strong reactions with bulk Na.

Thus, with hot WVN, a regular destruction of sodium and derivative is expected, with a good efficiency in term of consumption of injected water.

1.2.3. Third phase: sodium hydroxide hydration

The goal of this phase is to transform the anhydrous caustic soda, sole product nominally remaining in the cold trap after the previous phase, in aqueous caustic soda, which can be accepted by the liquid effluent processing station of the power plant.

For the moment, it has been chosen to do this at room temperature, to prevent Caustic Stress Cracking, because one aim is to dissolve the caustic soda inside the cold trap, which avoids using an auxiliary dissolution vessel that will be a supplementary scrap.

Next the cold trap is water washed by the mean of a series of water filling, circulation of the liquid content of the trap and then draining of this content. At last, the cold trap is dried under nitrogen flow.

1.3. Conclusion

Analyses of former experiments of cold trap processing and theoretical thoughts have led to select a process for the dismantling of SPX cold traps. This process will now be put on the test. In case of complete success, SPX cold traps dismantling will have marked a decisive step in this delicate problem of cold trap processing, which has until now not yet received a definitive answer.

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PHENIX PLANT DECOMMISSIONING PROJECT

M. SOLDAÏNI, M. DELUGE
CEA VALRHO, BAGNOLS-SUR-CEZE, FRANCE

G. RODRIGUEZ
CADARACHE COMMISSARIAT A L'ENERGIE ATOMIQUE (CEA), CADARACHE,
SAINT PAUL LEZ DURANCE, FRANCE

Abstract

This paper gives a general view of the Phénix reactor decommissioning schedule. It summarizes the main steps of end of operations (Cessation Définitive d'Exploitation: CDE) and dismantling phases. These two phases are described from the final shutdown planned in 2009 to the end of dismantling around 2024. During the first phase, operations consist mainly in removing fuel and other materials. Most of the treatment facilities for sodium and wastes are built during this phase. During the dismantling phase, operations consist mainly in treating the sodium and dismantling the reactor and the other nuclear facilities and equipments.

1. INTRODUCTION

Phénix is a fast breeder reactor developed at the end of sixties. Phénix has been under operation since 1973 and was connected to the grid in 1974. This a second generation prototype developed when Rapsodie (first generation of French FBR) was operating. Phénix is a 250 electrical MW power plant. During the first 20 years of operating, its main aim was the demonstration of viability for the FBR sodium cooling concept. Since the 1991 French law related to the radioactive wastes treatment and reduction (called 'loi Bataille'), Phénix became an irradiation tool for minor actinide transmutation demonstration. To reach this new schedule for 6 new operating cycles, it was necessary to renew the plant and to obtain an extension of the reactor life time of ten years. Many works were performed from 1999 to 2003 for a total amount of about 600 M€. Today, with a realistic expectation, the final shutdown is planned at the beginning of 2009.

2. STATUTORY PROCEDURE

In 2003, the safety authority defined a new application of the French law for the decommissioning of nuclear plant. For the CDE phase, after the final shutdown, it is not necessary to obtain a new safety authorisation to perform all the operations described in the operating safety report. It is possible to modify this safety report in the limit of the experience accumulated by the plant. It is only necessary for this phase to describe:

- All the operations planned;
- The organisation planned during the phase;
- The procedure for safety report modifications;
- An agreement to produce the final radioactive state at the end of the phase.

This period called in France 'Cessation Définitive d'Exploitation' (CDE: End of operating) is dedicated to fuel and hazardous materials removal. This is a sort of preparation phase before the dismantling phase. During the dismantling phase, the plant management becomes different compared to the previous one. For this phase a new safety report is needed to prepare a general authorisation for the whole phase. Statutory procedure is complex and the permission is obtained by a decree. For the statutory procedure, the safety report is completed by:

- Justification of the decommissioning scenario;
- General rules for supervision and maintenance;
- Modifications of the emergency plan;
- Environnemental impact study;
- Waste study.

3. APPLICATION FOR PHENIX

The CDE phase includes principally fuel and reactor components removal plus the dismantling of engine room equipments (turbo-alternator). The main operations during this phase are:

- Fissile elements evacuation and plant system simplifications:
  a. Core elements evacuation (300 fuel SubAssemblies, 200 steel SA, 500 shielding cylinders);
  b. Uranyle nitrate evacuation (+ eventual dismantling) coming from the neutron beam facility;
  c. Turbo alternator line dismantling;
  d. Steam generators dismantling;

- Construction of sodium treatment facilities:
  a. NOAH equipment construction for large volume of sodium processing (Industrial Risk-IR);
  b. ELA (Active) equipments construction to treat residual amount of active sodium (IR);
  c. ELI (Inactive) equipment construction to treat residual amount of inactive or secondary sodium (IR);
  d. New handling arm manufacturing (500 inaccessible radial shielding rods).

- Dismantling contracts preparation.

Some sodium treatment operations are already authorized during this phase: washing of fuel subassemblies in the dismantling element hot cell (CEI), and washing of components in 3 washing pits inside the handling building. For cleaning elements from secondary loops and steam generator, a new non radioactive sodium treatment facility is needed. This facility is called non radioactive washing cell (ELI) and beginning of operation is planned during the CDE phase. During the CDE phase, most of the sodium treatment and sodium waste facilities will be designed and constructed. Depending on the general schedule, the authorization for using them will be necessary during the CDE and/or the dismantling phase. The CDE phase will last about four years. It is the time needed to evacuate all fissile and fertile sub-assemblies, and more than the half of steel reflectors and carbon/steel shielding rods. The main operations during the dismantling phase are:

- Sodium treatment;
- Radioactive sodium wastes treatment from other facilities;
- Sodium loops removal;
- Sodium wastes treatment;
- Reactor vessel dismantling;
- Hot cells dismantling;
- Sodium treatment equipment dismantling;
- Buildings cleaning up;
- Plant downgrading.

In the case of Phénix dismantling, a special authorization is also needed to receive and to treat sodium radioactive wastes coming from other CEA facilities and past experiments. At first, CEA had planned to build a new plant for sodium radioactive wastes including Phénix and Superphénix wastes. In 2003, this project was stopped and the decision was made to treat all CEA radioactive sodium wastes during the Phénix dismantling phase.

4. DECOMMISSIONING COSTS

In 2003, a feasibility study mainly based upon the Superphénix dismantling scenario was carried out. It concluded of a cost of about 650 M€ associated with an uncertainty of – 50 M€ and + 350 M€ (Fig. 1).

This cost is composed with:

— Maintenance and operating (CEA staff);	275 M€
— Dismantling operations:	
- Primary circuit;	120 M€
- Secondary loops;	1.5 M€
- Reactor building;	12.5 M€
- Handling building;	27 M€
- Non radioactive equipments;	4.1 M€
— Sodium and sodium wastes treatment;	180 M€
— Operating assistance.	25 M€

External uncertainties are mainly considering underestimation in the waste radiological classification. Maintenance and operating costs are based on an optimistic staff decreasing curve. It is planned that, as soon as 2013, the staff will decrease from 260 to 150. Ten years later, the staff is estimated to decrease down to 50 people (Fig. 2).

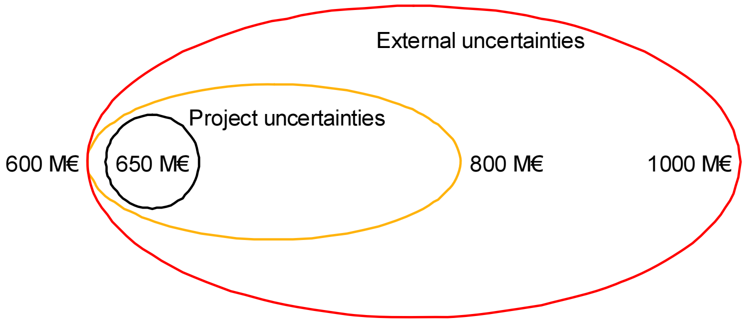

FIG. 1. Phénix decommissioning cost estimation.

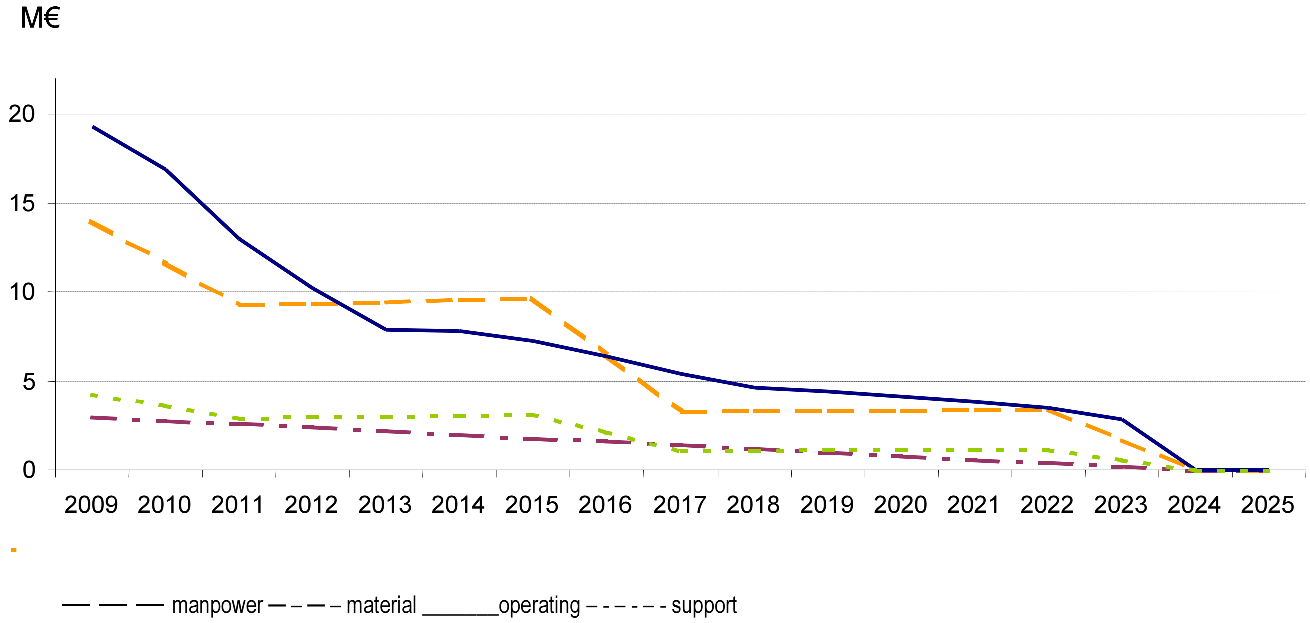

FIG. 2. Evolution of general costs and SENEX costs (Surveillance, Entretien Exploitation).

5. WASTES ESTIMATION

The estimated quantities of wastes:

—	Non radioactive wastes	3 000–5 000 tons
—	Very low level wastes	2 000 tons
—	Low level wastes	500 tons
—	Middle level wastes	250 tons
—	High level wastes	50 tons

Wastes destination depends mainly on the type of activity. Conventional wastes are transferred to storage centres (CET). Metallic pieces from the dismantling of non radioactive equipments are usually salvaged. For wastes coming from nuclear zones, depending on the contamination risk and history, every room is classified in:

- Zone generating non nuclear waste,
- Zone generating nuclear waste.

Every time a waste is generated inside the second type zone, the waste is at least classified as Very Low Level type, even is the activity is under the detection level. This is a specific French regulation. Therefore, for this type of wastes, a new national storage centre is operating since 2004. This centre is in the Aube department, few kilometres far from the Low Level Wastes storage centre (Centre de Stockage de l'Aube: CSA).

For Low Level wastes, after a first packaging at Phénix inside cans or metallic boxes, the wastes are repackaged and characterised in a central facility in Marcoule Centre before a road transfer to this French storage centre (CSA). Presently, there is no storage centre for ML and HL wastes. Different underground concepts are under evaluation before the creation of 'B' and 'C' waste types storage centre.

A new ML waste interim storage facility is under construction in Cadarache Centre and a new HA wastes interim storage facility is planned in Marcoule for the next five years.

6. PROJECT ORGANISATION

During the study phase, a CEA team will define the dismantling scenario. This team will prepare the two main reports for CDE and dismantling phase with the assistance of three main subcontactors. The first one in the field of wastes and transportation, the second one in the field of cleaning up and dismantling studies, and the third one is AREVA/FRAMATOME ANP in the field of Sodium Fast Reactor working and sodium treatment. For the last one, a specific department located in Cadarache is playing the role of expert for sodium handling and processing. The actual project is divided into ten sub projects as follows:

- Sodium treatment;
- Sodium waste treatment;
- Fuel removal;
- Component removal;
- Wastes;
- Reactor building;
- Conventional building;
- Handling building;
- Safety;
- Management.

7. END OF OPERATING PHASE

7.1. Fuel removal

All the fuel SA will be dismantled inside the fuel dismantling hot cell where the pins, after a washing process to clean all residual sodium, are separated from the structure. Pins are packaged into canisters and sent to La Hague Reprocessing plant. Structures are cut and put inside cans, then send in a specific facility in Marcoule centre called DIAM pit.

7.2. Component removal

All the facilities for component washing and dismantling are already under operation. This is mainly different casks, 3 washing pits and a special hot cell for components. After a liquid decontamination process, components are dismantled inside light vinyl cells in the handling building.

7.3. Sodium treatment

Different processes were developed for sodium treatment. After the Rapsodie accident, CEA concentrate its efforts on water/soda–sodium reaction. Several processes (more or less different in their basic principle) are selected for the treatment of all sodium and sodium waste coming from Phénix. The process selection depends mainly on the nature and the activity of the wastes concerned. Washing pits for components and core elements already exist, other equipments need to be constructed.

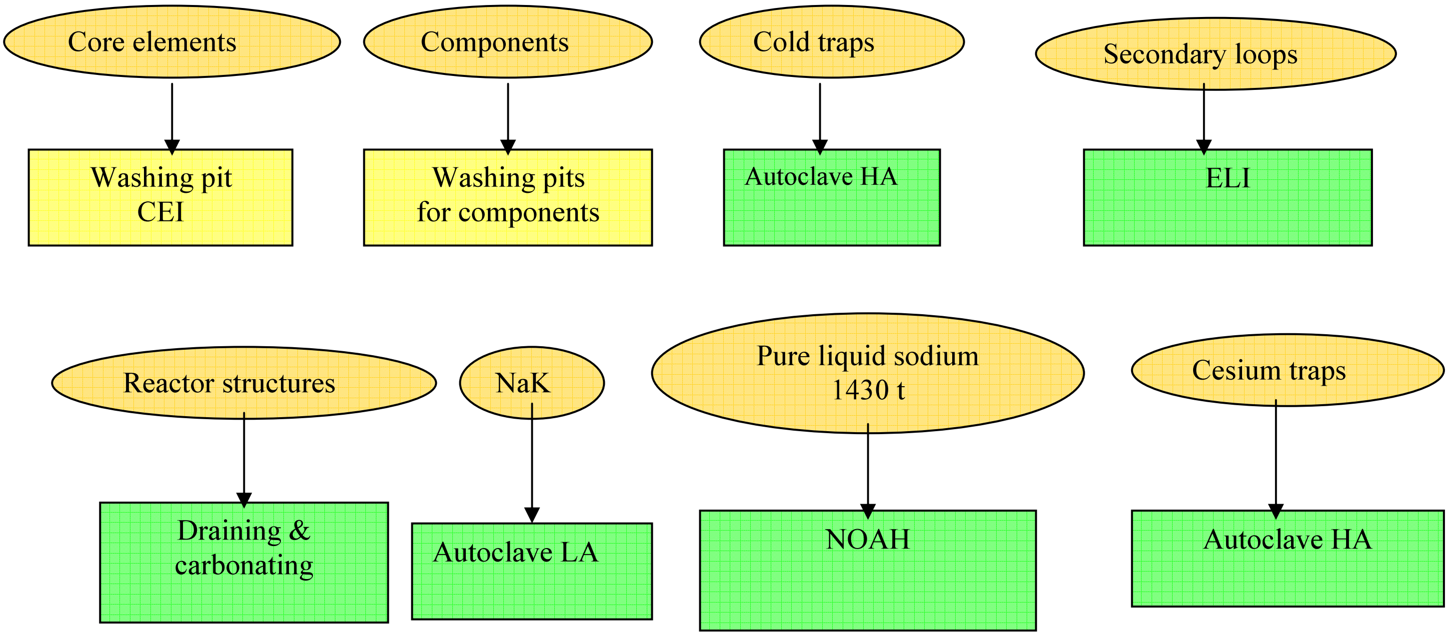

FIG. 3. Sodium waste treatment corresponding to several processes.

7.4. Core elements – washing processes

The core is composed with fissile and fertile subassemblies, reflector subassemblies and shielding rods containing graphite blocks. At the final shutdown, the number of sub assemblies is estimated to more than 300 fissile and fertile SA, 200 reflector SA, and more than 1000 neutronic protection rods. The dismantling route for all core elements is, after decay heat in the intermediate storage tank washing the residual sodium in the specific washing pit, cutting and packaging in accordance with the storage facility agreements. This type of dismantling operation has been carried out in Phénix since the beginning of the reactor operating. Therefore the defueling process is very well known. The washing process consists in the use of nitrogen and CO_2 humid gas before a complete water immersion.

7.5. Components – washing process

Because of some technical difficulties with the heat exchangers and with the primary pumps and the control rods mechanism, removal and dismantling of components have been also carried out since the beginning of the reactor operating. Three washing pits are in operation: one for big components and two for little components (small diameter) as measurement devices and control rod mechanisms. After washing and eventually decontaminating, components are repaired or dismantled in a special cell called CDI (Cellule de Démantèlement ou d'Invervention – Dismantling cell). The washing process consists in the use of nitrogen and CO_2 humid gas before a complete water immersion.

7.6. Cold traps – ELA process

Different processes were under competition for primary and secondary cold trap treatment. Because of the high activity of the primary cold trap (between 8 to 16 TBq in ^{137}Cs), the ELA process, developed for ATENA facility design is finally selected. The ELA process concerns a maximum amount of 40 tons sodium with a significant β/γ activity. Neither the less because of the low efficiency compared to NOAH process, it will be better to drain all pure sodium every time it is possible. The complete process is as follows:

- Preparation of the waste;
- Introduction in the active cell;
- Cut of tank and traps with a band saw or other specific cutting tools;
- Constitution of baskets with a capacity of 50 kg to one ton of sodium;
- Washing of the basket in the ELA reactor;
- Cutting and packaging the solid waste;
- Evacuating of liquid waste to the central facility.

This procedure consists in making a chemical reaction with sodium by water atomisation in a nitrogen flow. Injection by nozzles allows a local cooling and permits an intensive chemical gas sweeping in the reaction area. The water flow injection can be increased to reach the sodium in the top of the basket. The coverage, by a water shadow in an intensive nitrogen flow, reduces water retention fraction. Injection from the bottom makes the hydroxide sodium evacuation easier.

A recycling device (5 000 Nm/h) cools and purifies the containment atmosphere. This treatment process involves the production of:

- Gas with hydrogen and nitrogen (stripping, air dryer, filtration);
- Sodium hydroxide (from 0.5 to 10 Mol/L and variable specific activities);
- Metallic wastes without sodium.

7.7. Secondary loops – ELI process

Secondary sodium is usually contaminated by tritium at very low level. The process selected for the washing of pipes and components of the loops is close to the Cadarache facilities for non radioactive sodium. It consists in a cell of direct water aspersion on the sodium and wastes. This process is associated with a dismantling cell for packaging wastes.

7.8. Reactor structures – Draining and carbonation

After core elements removal, only diagrid and some internal structures very close to the core will be very highly activated. Before cutting probably under water these structures, primary sodium will be drained by pumping method. Residual sodium will be treated by carbonation before filling up the reactor vessel with water for biological protection during decommissioning operation.

Carbonation process needs some further developments to be sure that all different residual zones identified on Phénix reactor will be treated in depth.

7.9. NaK – Low activity autoclave

NaK inventory is around 2 tons. Most of the NaK will be mixed with primary or secondary sodium before treatment with NOAH process. For the residual NaK, the Autoclave process is needed. It consists in introducing a limited amount of sodium or NaK in a reactor vessel containing 50 L of water. A basket is introduced in a closed reactor filled with nitrogen. The complete process is as follows:

- Waste preparation;
- Introduction in the active glove box;
- Waste cutting with a band saw or other cutting tools;
- Introduction into the reactor;
- Cutting and packaging the solid waste;
- Evacuation of liquid waste to the central facility.

The reaction between water and sodium results in a raise of pressure and temperature. When these two parameters are stabilised, the reactor is decompressed and the gas is diluted after treatment (stripping, air dryer, and filtration) in an inert gas flow.

7.10. Caesium traps – High activity autoclave

Caesium traps are mainly coming from other CEA facilities. Activity of one trap can reach up to 10 Tbq of ^{137}Cs. The process is the same as for NaK, but the reactor will be located in the fuel element hot cell. Complete process is exactly the same as for NaK using an alternative saw for cutting the trap.

7.11. Primary and secondary sodium treatment – NOAH process

For pure liquid sodium, there is no doubt on the process selected. NOAH process was used successfully for Rapsodie and Dounreay sodium treatment and a NOAH equipment is under construction at Superphénix plant. The reaction is controlled by a specific chemical reactor cooled by the sodium hydroxide re-circulation and drain by overflow. Cut off of the tanks is made with a saw, before cleaning the metallic parts covered with residual sodium by ELA.

7.12. Dismantling phases

This phase is not yet enough defined to be presented in this paper. The dismantling scenario of the reactor is presently close to the Superphénix, one using the core element hot cell to package the HA wastes. Developments are planned with a laser for under water cutting.

8. CONCLUSION

Up to now, there is no real difficulty identified in the dismantling schedule because of the advanced state of development of the different processes selected for ATENA project. Because of the knowledge already obtained, stakes are more in project management and human resources reduction than in technical challenges.

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REACTOR DECOMMISSIONING STRATEGY

(Session 4)

DEVELOPMENT AND APPLICATION OF THE WATER VAPOUR NITROGEN (WVN) PROCESS FOR SODIUM RESIDUES REMOVAL AT THE PROTOTYPE FAST REACTOR, DOUNREAY

J.B. GUNN, L. MASON, W. HUSBAND, A.J. MACDONALD
UNITED KINGDOM ATOMIC ENERGY AUTHORITY, DOUNREAY, UNITED KINGDOM

M.R. SMITH
AMEC NNC LIMITED, BIRCHWOOD PARK, WARRINGTON, UNITED KINGDOM

Abstract

The decommissioning programme for the United Kingdom Atomic Energy Authority's two fast reactors at Dounreay is currently at the early implementation stage. UKAEA embarked upon an extensive development programme in 1994, to qualify the water vapour nitrogen (WVN) process for treating alkali metal residues. The aim of this programme was to identify a safe operational envelope that could be applied to a wide range of residue types and specific reactor geometries. Four stages of the development programme have been completed. The first stage was an extensive programme of approximately 100 small-scale tests. The second stage involved pilot scale tests up to 10 kg. The third stage involved the formation of an alliance of five companies and construction of an industrial pilot scale facility, for up to 1 000 kg, at Janetstown, ten miles from Dounreay. The fourth stage refined the process by cleaning actual sodium residues within various storage vessels and the secondary sodium circuits, with the aim of having a robust process for cleaning the reactor vessel. This paper summarizes the tests and the potential solutions too many of the issues identified during the course of the development programme. It also covers the lessons learnt from completing the cleaning of the storage vessels and secondary circuits. Finally, the next stage is also discussed. This covers the construction of a new on-site facility to clean vessels and plant items with tritiated sodium residues and further on-going development of the WVN process for cleaning the reactor vessel, in the quest to avoid unstable reactions.

1. INTRODUCTION

The United Kingdom Atomic Energy Authority's (UKAEA) Dounreay site in the North of Scotland is the location of the UK's two fast reactors; the Dounreay Fast Reactor (DFR) and the Prototype Fast Reactor (PFR) (see Fig. 1). Both reactors were cooled by alkali metals and are currently being decommissioned.

FIG. 1. The dome of the Dounreay Fast Reactor located between the Materials Test Reactor (left) and the Prototype Fast Reactor (right).

The DFR was the first fast reactor in the UK to supply electricity for public consumption and was commissioned in 1959. It is a loop type reactor cooled by a sodium potassium alloy (76:24 NaK). Due to the fuel being directly vented to the NaK and clad failures, it is highly contaminated, mainly with caesium 137. The reactor was shut down in 1977 and decommissioning commenced. Constraints in the fuel cycle area halted decommissioning activity in the early 1980's and approximately 57 tonnes of NaK remain in the primary circuit loops, together with 977 breeder elements in the outer core. The bulk of the coolant will be processed in a purpose built NaK disposal plant, which is shortly due to start commissioning. However, it is estimated that approximately 2 tonnes of residual NaK will remain distributed in the circuits on completion of this operation. This will need to be treated in situ and the reaction products removed, in order to reduce the radiation levels to the point where dismantling can commence.

The PFR is a pool type reactor cooled by sodium and was commissioned in 1974 and shutdown in 1994. Although much larger than DFR, containing an inventory of some 1 000 tonnes of sodium, the activity level is much lower due to the use of fully clad fuel with limited failures. The bulk sodium is currently being destroyed in a dedicated disposal plant and over 1 000 tonnes have been processed [1]. It is currently estimated that between 5 and 20 tonnes of residual sodium may remain in the reactor following bulk treatment. Although options to minimise the residues are still being considered it is likely that appreciable deposits will remain within the reactor vessel.

It had long been recognized by UKAEA that some form of in-situ alkali metal treatment process would therefore be required for potential application in both DFR and PFR. Following a review of a wide range of options, the method that has been selected is the Water Vapour Nitrogen (WVN) process.

2. THE WATER VAPOUR NITROGEN (WVN) PROCESS

The WVN process for sodium or NaK is represented by the following chemical reaction, Eq. (1):

$$2Na(K) + 2H_2O = 2Na(K)OH + H_2 + \text{heat} \qquad (1)$$

WVN thus represents one of many aqueous based processes that have been employed world wide to convert alkali metals to inert hydroxides. In this variant, the nitrogen performs the role of an inert carrier gas to transport water vapour to the alkali metal and excludes oxygen, thus avoiding the potential for hydrogen oxygen reactions.

UKAEA selected the WVN process on the grounds of controllability and suitability for cleaning large components at relatively low temperature. Although the work by Crippen in the USA [2] had established the basic parameters, there was evidence from previous industrial use of a number of outstanding issues. These primarily related to instability in the reaction rate and uncertainty in end point determination.

In order to qualify the technique for use at Dounreay, UKAEA initiated an extensive development programme in 1994. The primary aims were to establish a safe operational envelope and to explore specific issues associated with applying WVN to the wide range of deposit types and geometries associated with DFR and PFR.

3. DEVLOPMENT PROGRAMME

Due to lack of on-site facilities at Dounreay, UKAEA contracted NNC Ltd (now AMEC NNC Limited) to carry out the development work at their laboratories in Warrington, England. The initial work to establish a safe operational envelope was carried out using laboratory and pilot scale facilities. These facilities consisted of two purpose built, nitrogen inerted gloveboxes and three test rigs. Water vapour concentration, WVN flow rate and temperature were the main parameters varied in this exercise. Reliable measurement of hydrogen levels, accurate to parts per million levels, was achieved.

The initial development programme was broadly split into two stages. The first stage defined as laboratory scale tests, focussed on defining the reaction characteristics over an extensive range of operating parameters. The second stage defined as pilot scale tests, examined scale up of the process by examining the application of the process to sections of full size geometrical features and some generic areas such as pools [3].

3.1. Laboratory scale tests

The 100 laboratory scale tests established the reaction characteristics over a wide range of parameters. Temperatures in the range 40-120°C were investigated with water vapour concentration up to 15% and order of magnitude changes in flow rate. Each sample was approximately 12 grams and generally contained in 25 mm deep, 26 mm diameter stainless steel pots. Some additional geometries and deposit types such as crevices, deeper samples and frosts were also investigated. A limited number of tests were also carried out to investigate the effect of adding carbon dioxide or oxygen to the WVN mixture. For simplicity, sodium was used to establish the majority of the characteristics with only 11 NaK tests being carried out, to identify if there were any major differences. The main conclusions drawn from the programme were:

a) The optimum process condition lay in the temperature range 70-90°C, with water vapour concentration up to 5%. This selection gave acceptable reaction rates for most situations, was above the sodium hydroxide mono-hydrate boundary at 62.6°C, but below the melting point of sodium (98°C).
b) Operation at water vapour concentrations in excess of 10%, high velocity or where the sample was well insulated all increased the probability of unstable reaction rates.
c) The reaction rate was little affected by sample orientation, small changes in velocity or the presence of oxygen.
d) Addition of CO_2 increased the reaction rate. This had been originally postulated as a possible emergency shutdown mechanism but was found to be ineffective. It was found that the carbon dioxide reaction with sodium hydroxide resulted in the release of additional water via the reaction, Eq. (2). As this water is released the sodium hydroxide solution strength reduces. This in turn leads to an increase in vapour pressure at the sodium surface, an increase in reaction rate and hence temperature. The positive feedback of this mechanism results in a runaway reaction.
e) There were only small difference between sodium and NaK, which appeared to be mainly due to the fact that the relevant NaK mixture (76:24) is liquid in the processing temperature range. Surface tension may also have played a part due to the relatively small size of the samples.
f) Overall the process appeared to be safe and feasible for application at industrial levels, subject to carrying out further tests at larger scale.

$$2NaOH + CO_2 = Na_2CO_3 + H_2O \tag{2}$$

3.2. Pilot scale tests

The primary aims of this stage were to examine direct scale up of the process and investigate issues associated with applying the process to representative areas of the reactor geometries. Scale up was addressed by carrying out tests with bowl geometries up to 400 mm in diameter, containing NaK or sodium samples in the range 2–10 kilograms. For the largest test this represented an area increase by a factor of approximately 250. Other geometries studied in this phase included, as follows:

Geometries for DFR:

a) Sections of the core melt out tray. This comprises a number of petals shaped segments each 50 mm deep and located directly beneath the core region.
b) Sections of the primary circuit hot traps.
c) A long pipe, representative of the geometry of a single primary circuit heat exchanger loop.

Geometries for PFR:

a) A section of the low pressure plenum, which is an annular ring box structure around the base of the core, with a single entry point. The condition modelled assumed a partially drained geometry with a 10mm deep sodium layer.
b) A section representing the base of a drained heat exchanger. This also contained a 10 mm deep sodium layer.
c) An area near the top of the reactor incorporating several transition welds and which is likely to contain condensed sodium frost.
d) A section of a primary circuit cold trap.

The pilot scale tests confirmed the applicability of the process envelope, defined by the laboratory scale programme, for a large number of the above situations. In particular where the alkali metal residues were present as films or deposits, no real processing problems arose. Some of the tests did however highlight interesting issues, two are discussed below.

3.2.1. Sodium pool issues

The tests with 400 mm diameter bowls of sodium were the first to identify a major issue, associated with the processing of large horizontal deposits, upon which liquid hydroxide could accumulate to significant depths. This situation can lead to stall of the reaction process, as transport of water to the hydroxide/sodium interface is hindered by the increasing thickness of the hydroxide. If the WVN process is allowed to continue, dilution of the hydroxide occurs and subsequent perturbations, such as a small temperature change, can result in rapid restart of the reaction. The positive feed back effect on temperature provided by the exothermic nature of the reaction accelerates this process. The result can be rapid increases in pressure, temperature and hydrogen concentration, which is undesirable from a control point of view and can challenge system pressure boundaries and downstream off-gas plant.

This issue is particular relevant to PFR as one of the major residues is the heel at the bottom of the reactor vessel, which is currently estimated to be between 2 and 7 tonnes.

3.2.2. Primary cold trap loop issues

An option study was performed to establish the feasibility of using the WVN technique for cleaning the Primary Cold Trap Loop (PCTL) baskets in situ, in the PCTL vessel. The Cold Traps are highly active and contain a matrix of sodium and trapped impurities such as sodium oxide and hydride, held in the stainless steel mesh baskets that form the traps. One of the main recommendations from the study was the execution of a development programme to examine the performance of the WVN process on the PCTL deposits. The first aspect that needed to be addressed was confirmation that WVN treatment of the sodium oxide and hydride could be carried out successfully. Following this, establishing a suitable process envelope for WVN treatment of a matrix of sodium, oxide and hydride deposited in a steel mesh was required.

Four tests were performed in this programme of work, initially with small scale tests on sodium oxide, sodium hydride and a mixture of the oxide, hydride and sodium. These tests showed that there were no adverse effects during WVN treatment and the main problem would be the geometry of the baskets. The fourth test in the programme addressed this and a full size 30° sector of a cold trap filter was replicated. To represent the worst case, this was fully loaded with sodium. It took 83 days to completely clean the basket using continuous WVN injection to a final level of 10% by volume at the end of the test.

FIG. 2. The industrial scale test facility at Janetstown.

3.3. Industrial scale tests

3.3.1. Industrial scale test facility

The success of the laboratory scale tests led to the requirement for industrial scale tests. An alliance of the following five companies was formed in 2000, to develop the process and then clean all of the residues at PFR; UKAEA, Sinclair Knight Merz, Alstec, JGC Engineering & Technical Services Ltd. & Jacobs Babtie. To facilitate the industrial scale tests, a custom-built test facility was constructed at the Janetstown industrial estate on the outskirts of Thurso, some 10 miles from Dounreay. The Janetstown Off-Site Test (JOST) facility was designed, built and commissioned within 12 months, see Fig. 2. The facility consists of three key plant sections, the WVN generation skid, the reaction test vessel, and the waste treatment skid.

The WVN generation is carried out by injecting steam into a nitrogen stream, producing a WVN flow of known water concentration. The reaction test vessel is a 7 m^3 vessel with a pressure containment capability of 25 barg. The waste treatment skid filters the gaseous effluent to remove sodium hydroxide aerosol from the gas before it is discharged to atmosphere. The liquid effluent is pumped to containers and disposed of by a third party.

Initial investigation on the PFR primary vessel had indicated that the major concern was the large deposit of sodium that would remain in the reactor heel area; it is estimated that between 2 tonnes and 7 tonnes of sodium will remain in this area following removal of the bulk sodium. Therefore, the experimental programme was designed to focus on the treatment of sodium pools.

Particular attention was paid to ensure that adequate monitoring and instrumentation was provided. The test vessel is fitted with numerous thermocouples located throughout the internal volume of the vessel. Together with pressure transmitters, these thermocouples provide information on the progress of the sodium-water reaction and the magnitude of any excursion events. Hydrogen monitoring is provided in the exit gas stream to quantify hydrogen concentrations during normal operation and reaction excursion events. A video camera is located at a viewing port on the reaction vessel to provide visual information on the progress of the sodium-water reaction.

3.3.2. Industrial scale test programme

The test programme started with some small-scale tests, using sodium pools of 5 kg and 10 kg to gain an initial understanding of the process mechanisms and quantitative information on the magnitude of the excursion events. The quantity of sodium was then increased to 25 kg, and a series of experiments set out to investigate the effect of running the process at different operating conditions. Subsequently, following investigation of the different operating conditions, the magnitude of sodium used was increased to 100 kg and finally to 1 000 kg. A total of ten tests were completed. The main conclusions from the test programme were that pressure excursions occurred frequently and were much lower in magnitude than the worst case pool scenario [4]. Hence the treatment of large pools without the removal of hydroxide solution was considered manageable. The following key lessons have been learned from the industrial scale test programme:

- Sodium pools of up to 1 tonne can be treated by the WVN process in a safe and effective manner, without the need to remove the hydroxide during treatment.
- Temperature and WVN concentration had no significant effect on the magnitude of pressure transients during excursion events.
- Temperature and hydrogen concentration have been demonstrated to be good indicators of reaction progress and of detecting a cessation in the sodium-water reaction.
- To prevent the build up of excess water in the NaOH solution it is suggested that heat be applied periodically to provoke small excursions. This should have the added benefit of reducing the size of excursion events that can occur.
- If reaction stall does occur, the reaction can be re-started by the application of heat.
- Reaction behaviour above 100°C gives rise to more frequent excursions that are less severe in magnitude than below 100°C.
- Reducing the nitrogen inlet flow rate to improve the exit instrumentation sensitivity is a useful technique to assist in identifying the end point. Checking that no increase in hydrogen generation rate occurs when the system is heated is another technique.
- Increasing the WVN concentration when the hydrogen reading dropped to zero, was a useful technique in trying to ensure that the last deposits of sodium were reacted.

The results of the test programme were used as a basis for developing the plant to clean various vessels and systems described in section 4. Despite the success of the test programme, the large scale difference between the test vessel and the reactor vessel, has meant that obtaining approval of the safety case for containment of a worst case pressure pulse, will be very it difficult. The development programme for cleaning the reactor vessel has therefore taken a new route and this is described in section 5.2.

4. COMPLETED APPLICATION OF THE WVN PROCESS

4.1. Miscellaneous vessels and plant items

The JOST facility was used to further develop the WVN process by cleaning various inactive vessels and items of complicated geometry, such as valves and small lengths of pipework. The vessels were cleaned in-situ and the various items were placed inside a basket and cleaned in the reaction vessel. The miscellaneous items were cleaned successfully, but during the WVN treatment of a tank containing several tonnes of sodium, a large pressure excursion was triggered and caused damage to the tank.

The tank had been installed at an angle in order to facilitate draining of the sodium hydroxide solution to a pool at the low end of the tank. It was anticipated that this would make the removal of sodium hydroxide more straightforward. However, the WVN processing was run at a relatively high temperature and this resulted in sodium hydroxide setting in the extract line, meaning that sodium hydroxide removal was not achieved.

The incident occurred because operation was allowed to continue without sodium hydroxide being removed. The slope on the tank meant that an exposed sodium surface was available for reaction for a longer time than would be the case for a flat surface, and this resulted in a far greater amount of sodium hydroxide solution collecting. Eventually the sodium water reaction did stall and when WVN was bubbled into the sodium hydroxide layer to provide agitation, a violent re-start accompanied by a large pressure pulse occurred.

The incident served as a reminder that sodium cleaning operations carry an inherent degree of risk and underlined the fact that rigorous design and analysis are required, prior to determining and implementing cleaning strategies. It also emphasises that operating procedures need to be accurately followed during the cleaning process.

4.2. PFR sodium tank farm

The tank farm facility is located adjacent to the PFR reactor building and consists of a building housing ten large mild steel cylindrical tanks (3.2 m diameter × 18.3 m long). The tanks acted as storage vessels for supplying fresh sodium to the reactor primary and secondary systems and also as dump tanks when the secondary systems had to be emptied. Hence the sodium inventory is contaminated with tritium. Six of the tanks were drained and each contained up to 300 kg of sodium residues. The WVN process has been successfully applied to all six empty tanks and four have been cut up and dismantled, see Fig. 3. The cleaning process took around two weeks for each tank. The remaining two tanks are to be used for a new sodium inventory disposal plant, described in Section 5.1.

FIG. 3. Sodium tank being dismantled after sodium residue removal.

The WVN was delivered to the tanks by a skid mounted unit, where the temperature, flow rate and water concentration was set by the operators. The sodium hydroxide resulting from the process was sparged with steam, diluted and neutralized in-situ. The steam sparging raised the temperature, provided mixing and continued during the dilution stage. The tanks were connected to a neutralization skid mounted unit and the diluted sodium hydroxide solution was extracted to it. The pH was measured on-line and was used to control the injection of the sulphuric acid into the hydroxide stream. The solution was circulated back to the tank and the process continued until the pH value met the discharge limits of between five and nine.

The exhaust gas from the tanks were passed to a knock out box, where liquid droplets and large particles were removed, before passing through a dry HEPA filter, a flame arrestor and then discharged to atmosphere via the tank farm's own dedicated stack. The lessons learned from the cleaning campaign were:

a) The process results in a number of small pressure excursions accompanied by loud 'pops' and 'bangs'. This should be viewed as normal.
b) Emergency shutdowns due to high hydrogen were unnecessary and the process trip could be removed. The hydrogen level was in excess of 2% during most excursions.
c) Condensation of water and sodium hydroxide occurs in the ventilation lines and drainage points must be installed at low points.

d) Reaction pressure excursions cause vibrations in connecting ventilation pipework and can cause weld failure. The design of connecting systems must take account of such vibrations.

e) The WVN process sometimes doesn't passivate all of the sodium and heat should be applied to react any remaining resides. Water dilution must be taken with care and the design should ensure that the integrity of the system is sufficient to withstand any potential sodium-water pressure excursions.

f) To avoid overshoot, the pH neutralization system should have the capacity to switch to a low strength acid when nearing the neutral point.

4.3. Secondary sodium circuits

The secondary sodium circuit transferred heat from the primary sodium system circuit to the steam generating plant. The system consisted of three independent closed loops and the majority was located in a building adjacent to the reactor building. Each of the secondary loops connected a steam generator to two of the heat exchangers in the reactor vessel and consisted of the following components; the shell side of two intermediate heat exchangers, the shell side of a reheater, a superheater vessel and an evaporator vessel, an expansion tank, a sodium circulating pump, complex pipework and isolation valves, impurity monitoring equipment and a hydrogen detection loop.

Each circuit was cut and capped to isolate the intermediate heat exchangers, before the WVN gas was introduced at multiple positions near the low points of the system. The WVN flow path was very tortuous before it reached the exit point at a high position. The amount of sodium residues in each loop was estimated to be up to 200 kg.

The WVN was delivered in stages up to 6% concentration until no hydrogen was detected. This was then increased to near saturation levels to passivate any remaining residues. The circuit was then filled with water and this weak hydroxide liquor was extracted, circulated through a neutralization system and put back into the circuit. After suitable recirculation and neutralization, the liquor was transferred to the site's effluent treatment plant by bowser. Regular small excursions were observed, but these did not cause a discernible increase in the system's pressure. However, excursions will cause vibrations in connecting pipework and this should be accounted for in the design of the system.

This effect was illustrated during treatment of secondary sodium circuit cell No. 3, where two leaks of sodium hydroxide occurred. The process was stopped and an investigation conducted into what caused the leaks, and how to progress with the project. In the case of the first leak, a gasket failed. The second leak was from a crack in a 4 inch diameter pipe. It was concluded that the pipe was deformed before WVN cleaning commenced. The predicted reaction force occurring during WVN is not large enough to produce the force required for pipe deformation.

It is postulated that deformation occurred during reactor operations, possibly due to thermal expansion, a large bending moment, vibration, over-pressurization or compressive buckling and this gave rise to a crack in the pipe wall that did not break through its surface. Subsequently, the operation of the WVN process caused vibration of the pipework giving rise to crack growth, leading to eventual surface breaking and giving rise to a leak path.

Following the incident, the integrity of the system was restored and WVN treatment of cell 3 is now complete. All three secondary sodium circuits have been cleaned and one has been cut up and dismantled.

This incident highlighted that vibrations in unrestrained pipework in the WVN process can give rise to potentially serious consequences. It is therefore critical that a thorough inspection of extant plant is carried out prior to WVN treatment to identify any weak points, and that pipework is restrained to prevent excess movement and pipe stressing from occurring. Despite filling the system with water, dismantling of the steam heat exchangers (tube bundles) and circulation pumps revealed small pockets of unreacted sodium. Thus any dismantling risk assessments must take account of the probability of discovering unreacted sodium.

5. FUTURE APPLICATION OF THE WVN PROCESS

5.1. Sodium inventory disposal plant

Previous tests, operations and initial decommissioning activities at Dounreay have resulted in a large number of sodium contaminated vessels and plant equipment being placed in storage, waiting cleaning. The majority of these items are inactive or have tritiated sodium from the secondary sodium circuits. Some have chemical impurities of sodium hydrides and oxides.

PFR has two facilities for cleaning sodium residues from plant equipment. The Sodium Removal and Decontamination Facility is a large underground vessel (2.7 m diameter by 18 m in length), which uses steam lances to clean residues from large primary circuit items, such as charge machines and primary pumps. The items are transported to the vessel by a flask. The Steam Cleaning booth is a walk-in containment structure (3.2 m × 2.1 m × 2.6 m high) used to steam clean medium active items such as manipulators and valves. These facilities are not suitable to incorporate the WVN process for cleaning the large number of items in storage, described above.

A new facility to clean the items described above, has been designed and is presently being manufactured. The Sodium Inventory Disposal (SID) plant is being built within the existing building that contains the remaining sodium storage tanks (sodium tank farm). The SID plant is being located within the footprint of the dismantled tanks. The sodium tank farm is a separate building located just northern of the main reactor building and has its own ventilation stack. The gaseous discharge authorization for the stack only covers tritium and so the SID plant cleaning process is restricted to inactive or tritiated alkali metals.

The SID facility incorporates two operational processes. Area 1 will be used to clean vessels in-situ and area 2 will have a dedicated reaction vessel to accommodate miscellaneous items and plant equipment. The building has an original concrete bund to contain any spillages and this feature is maintained for its new use. This has caused access problems for transportation in and out of the building and ramps have been added. The facility has no liquid drainage system and so a bowser transfer system will be used.

Two of the original sodium tanks which have been cleaned by the WVN process, remain in position and they will be used for the SID operations. Their re-use helps to minimise costs. One tank will be used to contain the hydroxide or neutralized liquors and act as a hold tank for discharge to the bowser. The other tank will be part of the ventilation system and will act as a pressure buffer/damper, upstream of and protecting the filters, instrumentation and pipework.

The ventilation system includes duty and standby filters, plus a flame arrestor. Aerosols containing dissolved sodium hydroxide and radioisotopes are always present and have to be ‘knocked out’ in order to prevent failure of the dry HEPA filters. Previous experience has also highlighted the need for drainage connections at all low points in the ventilation system, to drain off water and hydroxide condensate.

Vessels will be transported into area 1 and a lance will be inserted to deliver the WVN at a concentration of up to 10% v/v and a temperature of approximately 50ºC. At the end of the process, the WVN concentration will be increased before a water wash takes place. The neutralization system will extract the resulting hydroxide liquor, inject sulphuric acid and circulate back to the vessel. This will continue until the pH value falls between 5 and 9. If the volumes are too high for the vessel, the large hold up tank can be used to circulate the liquors.

The reaction vessel within area 2 will be cylindrical in shape (1.8 m diameter, 4 m long) with dished ends and will sit horizontally on a steel frame. The designed maximum working pressure is 7.5 barg. One end of the vessel is removable and is connected by a bolted flange arrangement. Due to the size and mass of the flanged end plate (2.5 te), a support frame will be used to roll it back from the vessel for loading purposes. Various flanged openings are provided for the application of services, control and instrumentation, liquid extraction, gaseous effluent and viewing. Supports are provided internally for two trays, on which the items are placed and wired into position. Access is sufficient for a forklift truck to place the trays into position, see Fig. 4.

The reaction vessel is fitted with heaters to melt out any sodium before applying WVN.

The WVN generation skid is self contained. The steam is injected into the heated nitrogen flow to produce the desired percentage concentration. Recorded parameters include pressure, temperature, water concentration, flow rate, hydrogen and oxygen concentrations.

The inventory of items and vessels to be cleaned has an estimated total of 7.4 E10 Bq of tritium, with nearly half of that being associated with one vessel. The sodium inventory for the reaction vessel in area 2 will be limited to 850 kg per load, to keep the potential excursion pressure below the designed operating pressure rating of the vessel. The plant has a design life of 10 years and installation is currently on-going. Commissioning is due in early 2006.

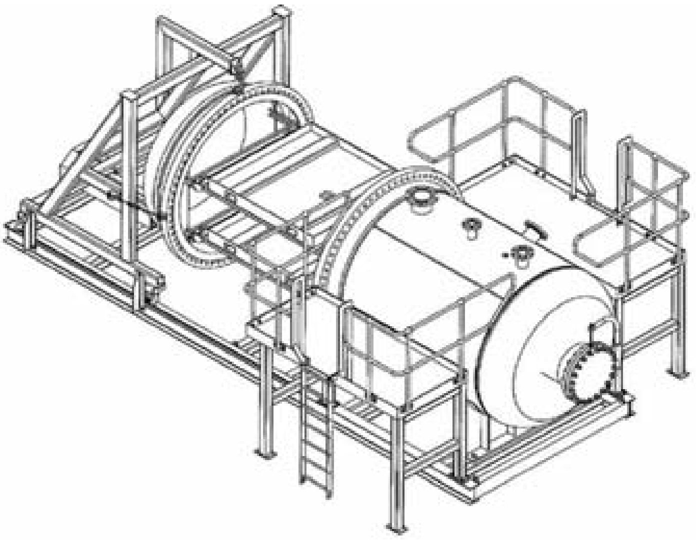

FIG. 4. A drawing of the sodium inventory disposal plant reaction vessel.

5.2. Reactor vessel

The complicated design of the reactor vessel internals means that obtaining access to remove liquid hydroxide from above the sodium layer is extremely difficult. The industrial scale trials at Janetstown showed that 1 te of sodium can be safely passivated using the WVN process, without the need for regular extraction of the hydroxide solution. However, the surface area of sodium at the bottom of the PFR reactor vessel is estimated to be up to nine times larger than the trials. This large gap between the scales makes it difficult to be totally confident that the process will not cause a pressure excursion which will give a breach of containment from the reactor vessel. The predicted pressure pulse from a worst case excursion has resulted in the need to justify the reactor vessel as a certified pressure vessel. This will be extremely difficult to undertake.

There are no available test vessels that can accommodate a sodium surface area approaching that of the reactor vessel. Computer modelling to increase confidence has been undertaken, but the many different and complicated factors of the chemical reaction, make it difficult to validate the model. Obtaining regulator approval of the safety justification, for using the WVN process without regular removal of the liquid hydroxide layer, may prove very difficult.

This factor, together with a window of opportunity due to programme changes, has resulted in a further programme of development work. The two aims of this recent task are:

— To increase the volume of sodium that can be drained for bulk disposal and so reduce the inventory for residues removal, in particular the size of the heel pool at the bottom of the vessel.
— To develop the WVN process, so that liquid hydroxide is not formed and so eliminates the potential for pressure excursions.

The major hold up areas for sodium is being re-investigated to see if the costs of technical solutions for their drainage are acceptable. Methods for physically checking the main level of sodium in the drained reactor vessel and the position of hold up areas are also being developed. Laser techniques and camera deployment are the main options being investigated.

In developing the process to form a dry reaction product, laboratory tests using low concentration WVN (<1% v/v water vapour) are underway. The process works, but initial results at low temperatures (50°C), indicate that it will take many years to passivate and reduce the level of sodium pools. The potential for operating at higher temperatures is being investigated.

6. CONCLUSIONS

The work reported in this paper forms part of the development of plant and process to achieve removal of residual sodium from UKAEA's Dounreay site. The paper illustrates the key elements in progressing the design, as well as the safety and environmental justifications.

The development strategy employed has been one of progressive challenge. Building on laboratory scale work, the experimental trials have scaled up the WVN process from 5 kg to 1 000 kg. This scale-up work has provided practical evidence of the efficacy of the WVN process in removing sodium residues. The trials have also been used to develop, test and refine the process plants required to generate the WVN and to treat the liquid and gaseous effluent produced. An important factor in achieving safe operation is to ensure that the process operators are suitably qualified and experienced and this has been achieved by training the operators as part of the experimental trials.

A sound process has been designed and has now been successfully used to clean various plant items. Problems arising along the way have been effectively dealt with and have highlighted issues to be accounted for in the design stage of future facilities.

However, the potential pressure transients that may be produced during excursion events make the safety justification for cleaning the reactor vessel very difficult. Therefore, the project has taken the decision to try to avoid the excursion events altogether, by developing a modified version of the WVN process, where the WVN is applied at low concentrations to produce a solid sodium hydroxide product. Development trials for low concentration WVN are in progress and the success of these trials will dictate the subsequent process methodology for sodium residue removal from the PFR primary vessel. Resources are also being deployed on methods to further reduce the size of any sodium pools in the reactor vessel.

The development programme has proved extremely valuable in identifying the key issues associated with applying the WVN process and was used in the design on the sodium inventory disposal plant. Whilst further development work is still required for the cleaning of large sodium pools, the work described in this paper has greatly increased the knowledge of the sodium-water reaction and has helped to ensure that sodium residue removal will be carried out in a demonstrably safe manner.

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CLOSURE OF THE FAST FLUX TEST FACILITY: HISTORY, STATUS, FUTURE PLANS

O.A. FARABEE
US DEPARTMENT OF ENERGY, RICHLAND OPERATIONS, RICHLAND, WASHINGTON, UNITED STATES OF AMERICA

W.R. CHURCH
FAST FLUX TEST FACILITY CLOSURE PROJECT, FLUOR HANFORD INC.,
RICHLAND, WASHINGTON, UNITED STATES OF AMERICA

Abstract

In 1993, the US Department of Energy (DOE) decided to shut down the Fast Flux Test Facility (FFTF) due to lack of national missions that justified the annual operating budget of approximately $88M/year. The initial vision was to ‘deactivate’ the facility to an industrially and radiologically safe condition to allow long-term, minimal surveillance storage until approximately 2045. This approach would minimize near term cash flow and allow the radioactive decay of activated components. The final decontamination and decommissioning (D&D) would then be performed using then-current methodology in a safe and efficient manner. The philosophy has now changed to close coupling the initial deactivation with final D&D. This paper presents the status of the facility and focuses on the future challenge of sodium removal.

1. INTRODUCTION

The FFTF is a 3-loop design liquid metal fast reactor rated at 400 MW(t) with no electrical power generation capability. The status as of August 2005 is as follows:

- Primary and secondary systems drained and cooled to ambient;
- Two spent fuel pools still filled with ~ 204 kg of ~ 400°F sodium;
- All ‘reactor’ related systems shutdown, most support systems active;
- 210 personnel onsite working on project;
- 37 of 376 fuel assemblies left to wash and remove from site.

At the current time, the end state of the site is unknown. The end state will be determined by a legally required environmental review process with extensive public participation. The alternatives range from entombment to site restoration. The entombment option would raze all buildings and structures to ground level, leaving the reactor vessel and internals in place, the below grade volume of the reactor building filled with grout, and a suitable environmental cover placed over the site. The restoration option would essentially return the site to a ‘green field’ status. For detailed planning purposes, the entombment option is being used. For funding purposes, the restoration option is being kept open.

The DOE is in the process of evaluating bid/tender proposals for the remaining workscope of the closure of FFTF within the following constraints:

- End State Entombment;
- Funding $46/M per year;
- Project completion 2012;
- Tender availability limited to small business (<500 personnel).

Major technical issues that need to be addressed include:

- Determination of end state;
- Method of disposition of ~ 1 134 kg of metallic sodium;
- Disposition of highly radioactive sodium containing components (traps);
- Method of disposition of sodium residuals following completion of bulk sodium drain.

The remainder of this paper will present experience to date in fuel washing, sodium drain and reaction, and the current strategy on resolution of technical issues.

2. BULK SODIUM DRAIN

2.1 Secondary sodium system drain

The three secondary loops were drained from April 2003 to September 2003. Since the secondary loops are designed for a rapid drain in case of leak, the drains were relatively simple and without incident. However draining of the Intermediate Heat Exchangers required insertion of 'dip tubes.' A hydraulic profile of a secondary loop is shown in Fig. 1.

2.2. Primary sodium system drain

The primary sodium system loops (Fig. 2) were sequentially drained to the Sodium Storage Facility leaving the sodium residuals as shown below in a representative single loop. There is no designed capability to drain the residuals shown in the main isolation and check valves. The sodium could have been drained to the top of the cold leg inlet piping by siphoning. This however would have left approximately 60 kl of sodium in the reactor vessel with no means of providing thermal heat input to maintain the sodium in the liquid state. The level in the reactor vessel was left at the level shown to allow thermal heat input from the installed immersion heaters which were located at the level of the core.

The trapped volume of ~ 60 kl that would have remained is shown in the Fig. 3. A hole was drilled into the lower plenum plate as shown; a dip tube inserted through the reactor head and core area, and the remaining sodium was pumped out using a fluidic pump.

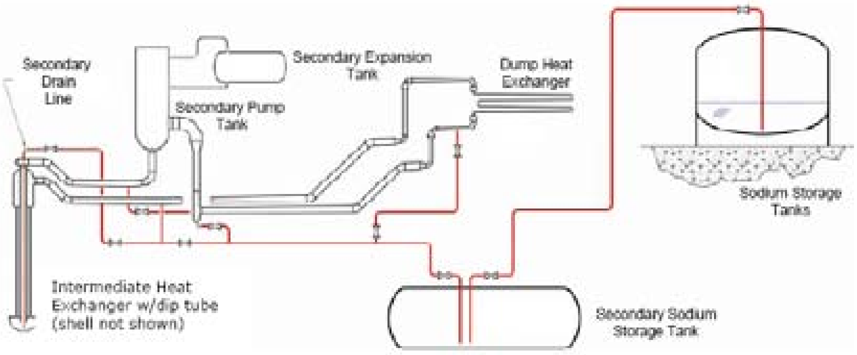


FIG. 1. Hydraulic profile of a secondary loop.

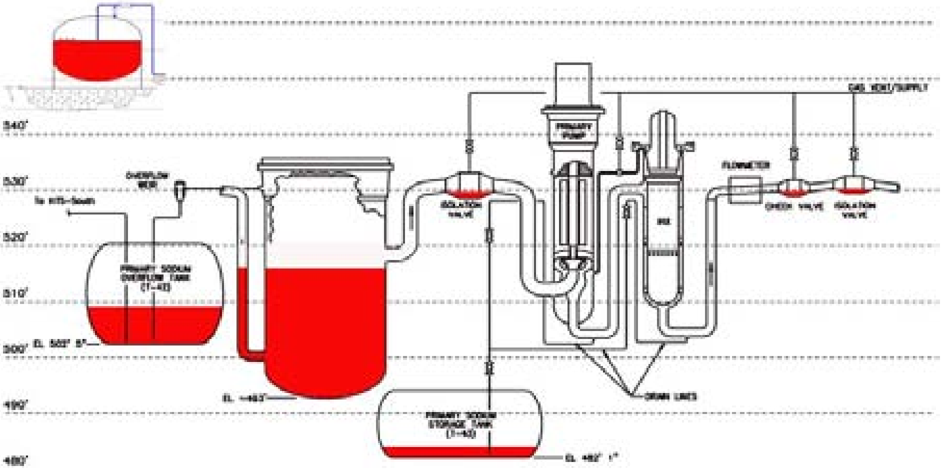


FIG. 2. Primary sodium hydraulic profile.

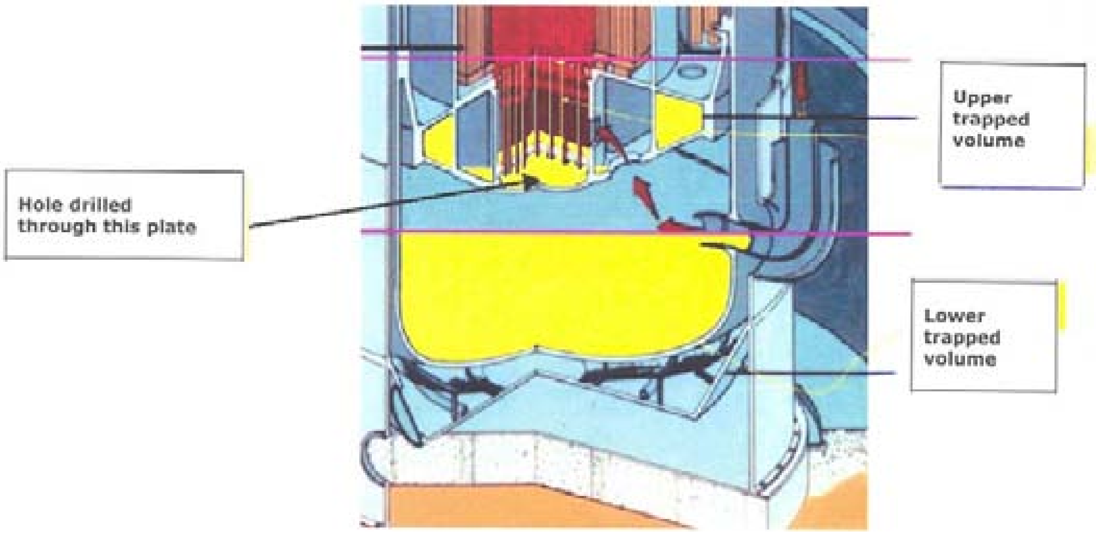
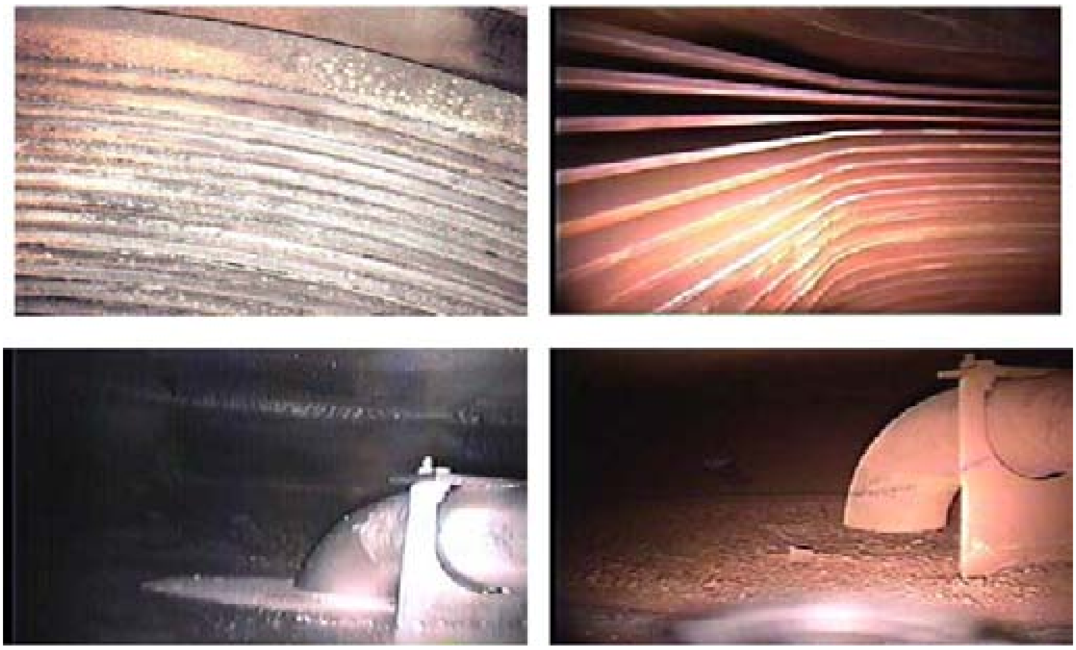


FIG. 3. Trapped (non-drainable) volumes.

3. CORE MOCKUP CLEANING EXPERIENCE

During initial design and testing of the FFTF, a one-third core mockup was built to test the in-vessel handling machines and instrument trees in a sodium environment. That vessel was drained in 1978 with only residual sodium remaining under a nitrogen blanket. Based on records and video inspection, it was believed that as much as 1 340 L of sodium remained in the vessel. A series of metal thermal baffles were located at the top of the vessel onto which it was believed that a significant amount of sodium frost had accumulated. Shown in Fig. 4 are photographs of the bottom of the vessel and the thermal baffle plates at the top of the vessel: ‘before cleaning’ (a) and ‘after cleaning’ (b).



(a) before cleaning *(b) after cleaning*

FIG. 4. Upper thermal baffles and bottom of tank.

This vessel was cleaned using the 'superheated steam' process. Based on chemical analysis of the final solution and based on the quantity of hydrogen generated, a total of approximately 350 L of sodium was removed as opposed to the initial estimate of 1 340 L being present in the tank.

The process was begun by heating the entire vessel and internals to ~ 200°C such that the sodium was fully molten at the start of the reaction process. The oxygen concentration was maintained at less than 1% and the tank pressure less than .034 atm throughout the process. The actual reaction time was ~ 7.5 hours but the flow of superheated steam and nitrogen was continued for ~ 100 hours at 82 kg/h and 17 m^3/h, respectively. The hydrogen concentration vented to the atmosphere during the 7.5 hour period varied from 30-36%. There was no apparent additional sodium reaction from the thermal baffle area when the steam flow was applied directly through spray nozzles, indicating that the reaction had been carried to completion as the steam from the bottom of the vessel had flowed through the baffle area. Following completion of the reaction the caustic solution, approximately 1 200 L was drained and the vessel was rinsed with water three times (including one complete fill cycle).

4. PLANS FOR SODIUM RESIDUAL REMOVAL

The current plans for the residual sodium removal from the FFTF systems is a combination of in-situ treatment by superheated steam and component removal and subsequent treatment by superheated steam. The major sodium wetted components are listed below and selected components are discussed in detail, and are listed below:

- Reactor vessel;
- Primary and secondary piping and tanks;
- Small bore piping (~ 4 115 m);
- Small heat exchangers and electromagnetic pumps (~ 33 223 m of tubing);
- Small sodium valves (~ 330) (may need to cut apart);
- Secondary cold traps;
- Primary cold traps, cesium traps and vapor traps;
- Fuel Storage Facility (FSF) and Interim Decay Storage (IDS) vessels and contents;
- Dump heat exchanger tube bundles;
- Primary/secondary pumps;
- Intermediate heat exchanger tube bundles;
- Sodium reactor experiment and hallam sodium containers (from other reactors).

4.1. Reactor vessel

The reactor vessel estimated sodium residual hold-up is ~ 750 L. Unlike the closure programs of other liquid metal reactors, the FFTF reactor will not be disassembled following cleaning. The vessel will be flushed with water to assure maximum reaction of metallic sodium. The only disassembly currently planned is the raising or removal of the instrument trees to access the thermal baffle areas (see Fig. 4). This is considered necessary because of a sleeve in the reactor head that segregates the head thermal baffles from the instrument tree thermal baffles.

4.2. Small bore piping, dump heat exchanger tube bundles, small heat exchangers, electromagnetic pumps and small valves

This group of components comprises a major portion of the sodium residual removal effort. These items and components will be removed from the plant and washed in an adjacent facility. The removal of the primary small bore piping will present the greatest challenge. The removal of each section of pipe will consist of:

- Appropriate planning and isolation;
- Sequential removal of stainless steel insulation cover, trace heat and thermocouples;
- Cutting and removal of pipe.

This work must be done in a radiation area and with all of the protective measures required for working with inert gas and metallic sodium. A diagram of the Dump Heat Exchanger tube bundle is shown in Fig. 5.

4.3. Sodium traps

The photo and diagram of the secondary cold trap is shown below in Fig. 4. These traps are currently full of frozen sodium and still installed in the secondary system. The traps will be cut out of the system, removed to an adjacent facility, drained and cleaned.

Due to the radioactivity level of the primary cold trap and the cesium trap, those components will be treated at a separate facility. Drawings of these two components are shown in Figs 7 and 8, respectively.

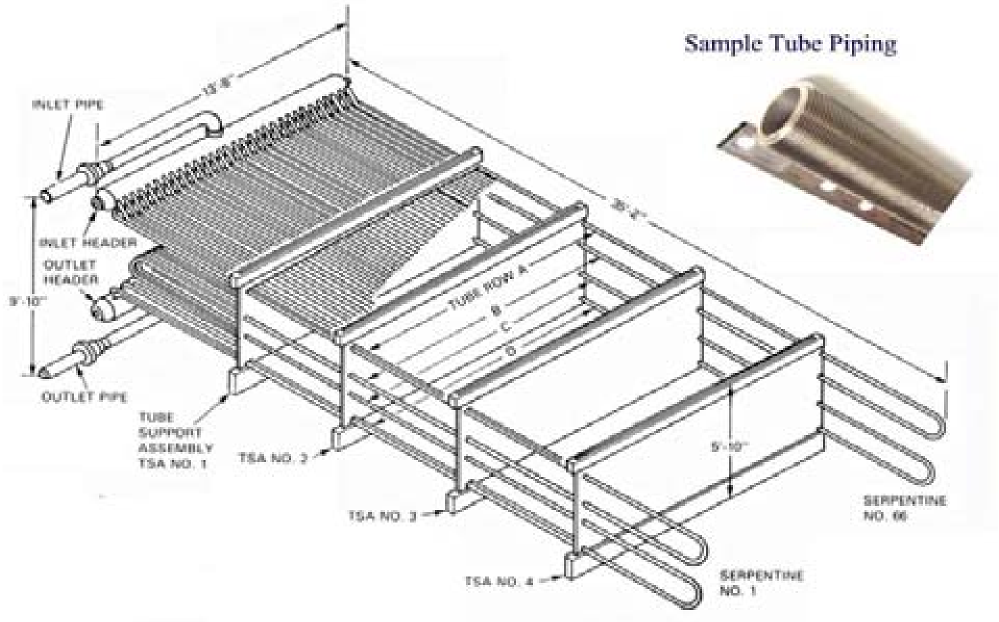


FIG. 5. Dump heat exchanger tube bundle.

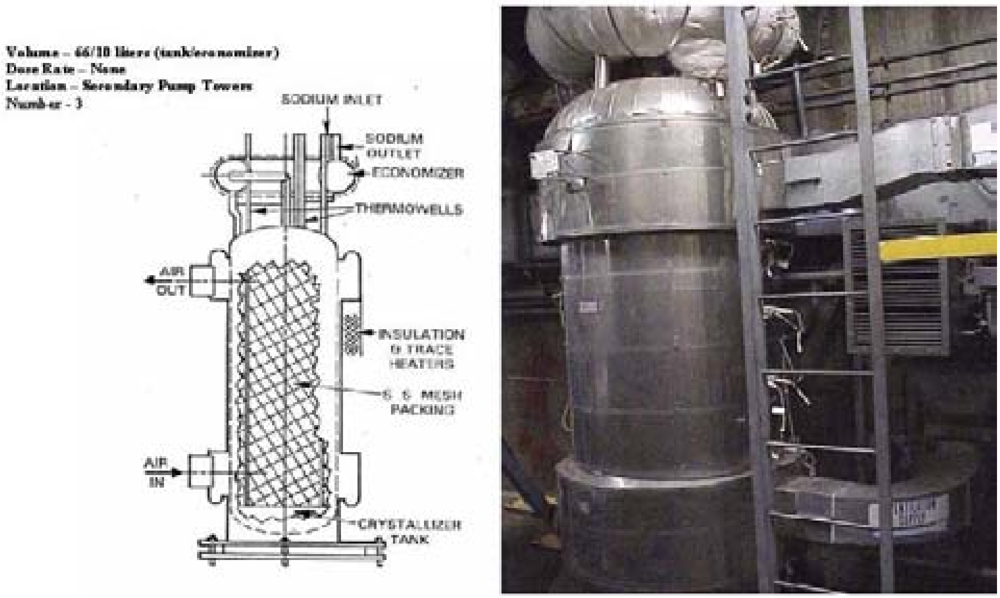
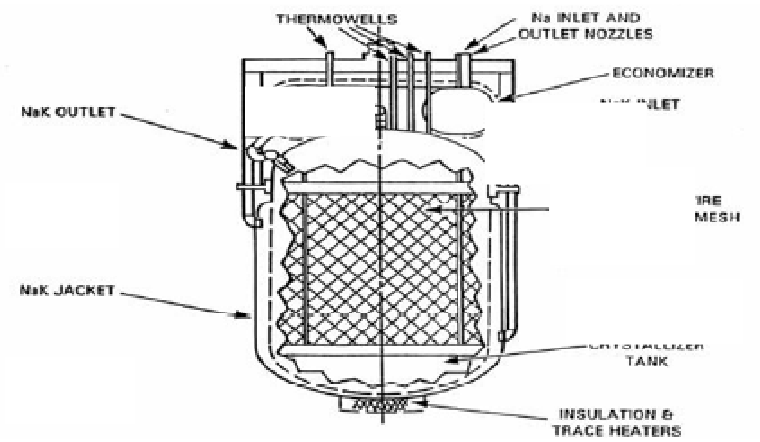


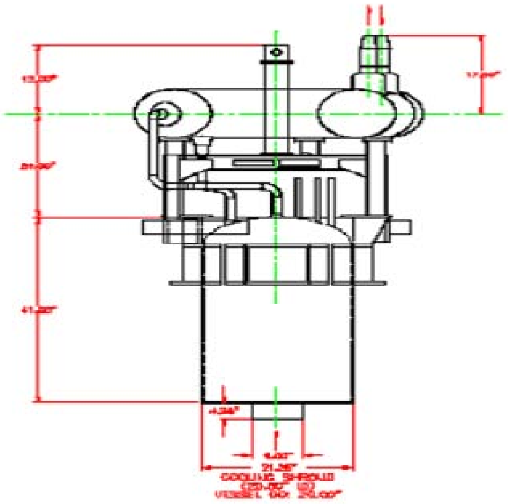
FIG. 6. Secondary cold trap.



Primary cold trap:

Volume: 170/15 L (tank/economizer)
Dose rate: 0.1-0.5 Sv/h
Location: Reactor containment
Number: 1

FIG. 7. Primary cold trap.



Cesium cold trap:

Dose rate: ~ 0.5 Sv/h

FIG. 8. Cesium cold trap.

4.4. Spent fuel pools (IDS and FSF vessels)

Each of these two vessels contains significant residual volumes. The IDS vessel contains 112 fuel storage tubes that will contain ~ 10-15 L each. The FSF vessel contains 466 tubes that will retain ~ 5 L each in the bottom of each tube. Both vessels have an inner sleeve which may require the drilling of steam injection points through pressure vessel for residual removal. IDS and FSF are shown in Figs 9 and 10, respectively.

4.5. Intermediate heat exchanger (IHX) and primary pump

The intermediate heat exchanger (IHX) tube bundle and the primary pump (Fig. 11) are the two largest components that will be removed for cleaning. The tanks or shells of these components will remain in the reactor containment building for residual sodium cleaning. The components are shown below installed in the tanks.

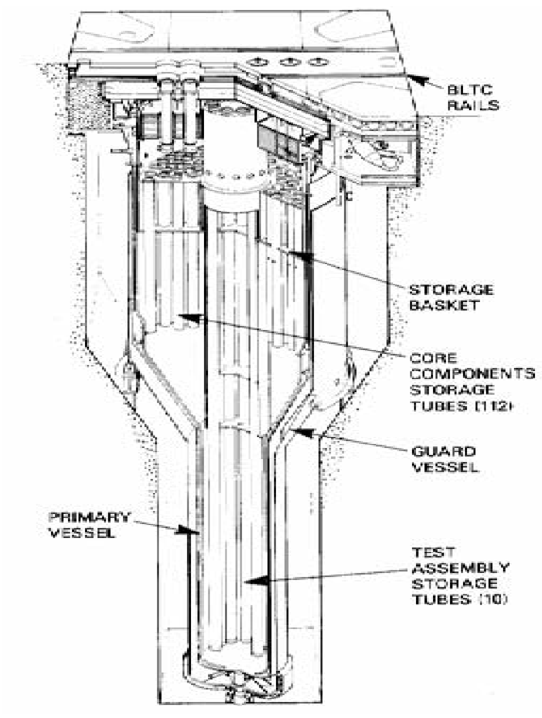


FIG. 9. IDS vessel.

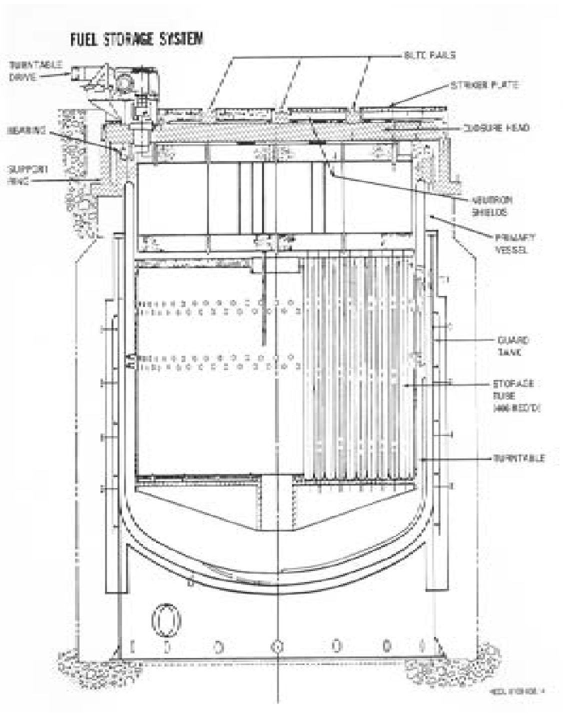


FIG. 10. FSF vessel.

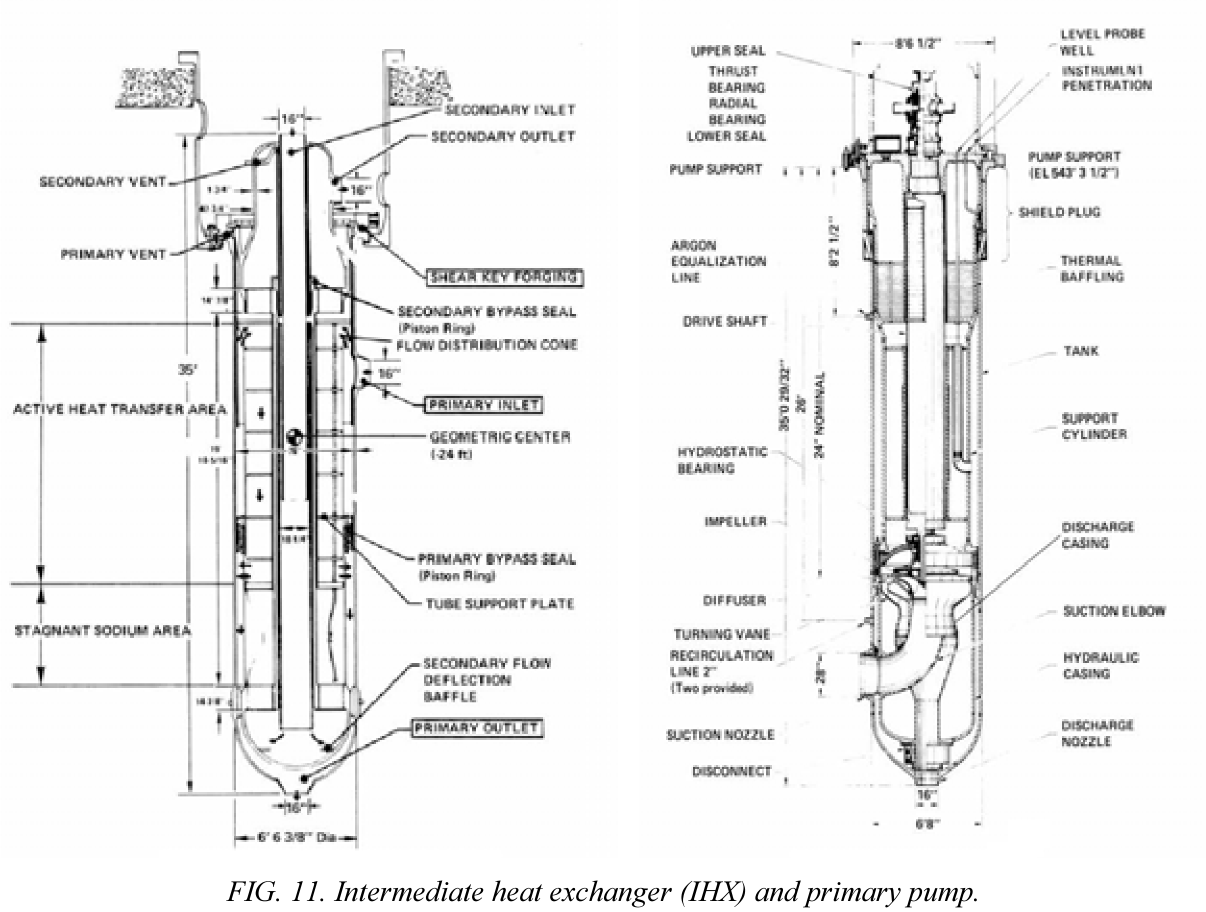


FIG. 11. Intermediate heat exchanger (IHX) and primary pump.

SUPERPHENIX – CREYS MALVILLE STRATEGY FOR DISMANTLING THE REACTOR BLOCK

E. JOULIA
ELECTRICITE DE FRANCE (EDF) – CIDEN, CREYS MALVILLE, FRANCE

Abstract

This document describes the operations to be performed to dismantle the SPX reactor vessel. The first studies for dismantling the Superphenix power station reactor block have defined the broad outlines of the dismantling strategy and proposed a certain number of orientations. The period covers SPX reactor block dismantling, from the end of fuel unloading until complete dismantling of the installations. Two dismantling methods have been considered: Underwater dismantling (reference scenario); Dismantling with the structures covered with sodium carbonate (alternative scenario). In both cases, dismantling will be carried out 'from the top' through the reactor slab and will be preceded by a preliminary phase of sodium retention reduction followed by a phase of vessel internals carbonation. The dismantling studies are undertaken with a view to the work lasting eight years maximum. The work prior to reactor dismantling will last from the end of 2011, after sodium draining from the reactor, until the end of 2013. At present, the Superphenix dismantling work is planned to begin in 2014 and continue for a period of eight years. All the installations should have been dismantled by 2025.

1. INTRODUCTION

After a large scope of preliminary studies carried out from 1999 to 2002, two dismantling methods have been selected:

- Underwater dismantling (reference scenario);
- Dismantling with the structures covered with sodium carbonate (alternative scenario).

EDF made a choice based on the following principles:

- Anticipation of dismantling operations in order to facilitate reactor dismantling;
- Reduction of the risks as early as possible;
- Closest possible containment;
- Operations carried out on the basis of an ALARA approach.

2. DESCRIPTION OF DISMANTLING METHODS SELECTION

2.1. Early dismantling operations to facilitate reactor dismantling

After preliminary EDF studies, it was proposed to extract the steel assemblies and dismantle all the removable components crossing the slab in order to facilitate dismantling of the reactor block. The presence of retentions in the base of steel assemblies and under the components, could exclude underwater dismantling, for reasons of acceptability in terms of safety.

The components will be replaced by plugs:

- Restoring the tightness of the slab containment barrier;
- Ensuring biological shielding of the slab zone after sodium draining.

Before beginning reactor dismantling:

- All its components will have been removed from the slab (except for the core cover plug and the rotating plugs);
- The steel assemblies will be extracted from the vessel.

2.2. Earliest possible reduction of risks

The sources of risks are:

- The presence of intense radioactive sources mainly consisting of certain radioactive reactor structures;
- The presence of metallic sodium on the internal structures.

As far as possible, the following risk sources will be eliminated at the earliest possible date:

- Metallic sodium;
- Radioactive sources.

2.2.1. Presence of intense radioactive sources

The presence of very radioactive structures is the source of difficulties for dismantling operations:

- Need to carry out certain operations by remote means;
- Risk of personnel radiation;
- Waste from dismantling the reactor structures attributed to different radiological categories;
- Difficulties of handling and packaging the most radioactive waste.

The result was to make recommendations for the elimination of the highly irradiated structures as soon as possible, i.e. right at the start of the dismantling operation (insofar as this is technically feasible), that is to say, to implement provisions aimed at assured reductions of the ambient dose rate. Therefore, the recommendations are:

- The most irradiant structures will be eliminated as early as possible;
- In view of the radiological conditions in the vessel after sodium draining, the operations will be performed by remotely operated equipment, at least until the environmental conditions authorize direct human interventions.

2.2.2. Presence of metallic sodium on the internal structures and the main reactor vessel

2.2.2.1. Reduction of metallic sodium related risk

Metallic sodium bas certain disadvantages: it is highly reactive with water and with other bodies, reaction products such as caustic soda and hydrogen can themselves be sources of risks. There are several methods of neutralization:

- Neutralization by oxidation;
- Neutralization by reaction with liquid water or a body associated with water;
- Neutralization by a mixture of carbon dioxide and steam that leads to the formation of sodium carbonate.

The carbonates obtained can be placed in a water solution during a washing operation. The process of carbonation by a mixture of nitrogen/carbon dioxide/steam is well documented, as well as the influence of the various parameters ([CO_2], [H_2O], T, treatment gas flow) on the nature of the bodies obtained, and on the kinetics: these parameters and their impact was determined by the 'CARNAC' tests carried out by the CEA at Cadarache;

- EDF have a feed back of SPX drum tank treatment by carbonation process;
- Exchanges with EBR II on Laboratory test, secondary circuits and primary vessel carbonation treatment.

After draining the reactor block, the residual metallic sodium is neutralized by carbonation. Total neutralization of the sodium cannot be guaranteed: metallic sodium may remain under the layer of carbonate and has to be taken into account for dismantling operations.

Carbonation of the sodium films should not present any particular problem.

Thickness of the sodium that can be 'carbonated in a reasonable time (a few months)' probably has limits: it will be difficult to neutralize the sodium on all the retention thickness if this exceeds a few centimeters due to the formation of a layer of carbonates covering the metallic sodium and preventing the uniform diffusion of the carbonation gas. In order to neutralize the largest possible quantity of sodium:

— The treatment gas must be able to reach the metallic sodium retentions;
— The treatment gas must be distributed as uniformly as possible in the vessel;
— There is as little retention as possible and these are as thin as possible.

After draining of the reactor block and before the residual sodium carbonation operation, the retentions will be treated in order to eliminate them or reduce the thickness.

2.2.2.2. Treatment of retentions

During 2000, it was proposed to treat (by drilling or siphoning) the five main retentions before draining the vessel, and – at the end of 2001 – to withdraw the steel assemblies and all the removable components crossing the slab.

Given that the removal of the primary pumps win give access to certain retentions, it appears convenient to examine the merits of treating them. In particular, the 'feasibility of demonstrating safety' aspect will be incorporated in the reflection as well as the technical and the cost aspects. The retentions will be treated as thoroughly as technical and economic considerations permit, taking into account the method of dismantling adopted and the associated safety studies.

2.3. Best achievable containment

During reactor vessel dismantling operations, the containment boundary will consist of the main vessel and the underside of the slab. The containment would then consist of the plugs sealing the penetrations left after extraction of the removable components, or airlocks for the transfers into and out of the vessel.

2.4. The ALARA approach

Dismantling operations will be performed using remotely operated equipment due to the dose rates encountered in the vessel. Direct human interventions will only be considered after withdrawal of the most irradiating structures or after implementation of measures to reduce radiation (removable protections or immersion of structures under water). The dosimetry targets will be defined and may require reviewing of the equipment used. In the field of radiation protection, all the dismantling operations will be carried out based on an ALARA approach.

3. ORIENTATIONS

The following decisions have been taken based on the ideas put forward and the dismantling methods selected.

3.1. Rejection of dismantling by the bottom

This would be a delicate operation as the components are situated between 2 and 20 m above floor level. If dismantling via the bottom is decided, i.e. transfer of the containment to the wall of the vessel pit. This increases the risk of carbonate dispersal to very probable when dismantling with sodium carbonate and completely excludes underwater dismantling. Lastly, dismantling by the bottom will not eliminate the most irradiating structures at the earliest stage, with the result that the operations become more delicate to execute. It was thus decided to reject this method.

3.2. Operations prior to the two dismantling methods envisaged

Whichever scenario is adopted out of the two dismantling methods considered, the operations will have to be preceded by carbonation of the sodium remaining in the vessel. This sodium neutralization phase is necessary to reduce the risk.

This phase consists of several operations which could be as follows:

- Inspection of the vessel internal structures after the draining operation;
- Carbonation to optimize the parameters based on preset targets (nature of the carbonates required, carbonation time limit, etc);
- Inspection of the vessel internals in order to check the efficiency of this operation;
- Additional treatments if necessary, according to the selected dismantling scenario.

Whichever scenario is envisaged it may be technically preferable to perform thorough carbonation. It will be remembered that:

- In the case of the underwater dismantling scenario, the reduced quantities of sodium remaining under the carbonate will facilitate filling of the vessel with water and reduce the safety constraints;
- In the case of the dry dismantling scenario, any reduction in the quantity of metallic sodium remaining under the carbonates will reduce the precautions to be taken during dismantling operations.

3.2.1. Treatment of the retentions

After draining of the reactor vessel ready for sodium treatment, elimination of the steel assemblies and the components crossing the slab, and without other actions than the elimination of the five main retentions, the estimated quantities of sodium are:

Wetted surfaces (films) $\approx 1.3\ m^3$

Retentions $\approx 1.3\ m^3$

The first observation is that the sodium coating will be completely eliminated during the carbonation operation, provided that the carbonation fluid is circulated uniformly in the vessel. Removing the reactor coolant pumps provides the opportunity for treating certain retentions which would not be accessible otherwise. If we only treat the retentions that are directly accessible after withdrawal of the reactor coolant pumps, it is estimated that a further 600 L of sodium could be eliminated. This would leave only approximately 700 L of sodium in retentions in the vessel (before carbonation).

Retentions $\approx 0.7\ m^3$

At present, we are able to guarantee that 10 mm of sodium would be neutralized by carbonation (provided that it takes place in the best possible conditions, i.e the retentions are accessible to the treatment gas and this gas can be renewed). However, the CEA glovebox tests have shown that several tens of millimeters of sodium can be at least carbonated in laboratory conditions. Therefore we have set a target which is considered ‘reasonably achievable’ and which is to carbonate 20 mm of sodium. On the basis of these assumptions, between 100 and 200 L of metallic sodium will be left under the carbonates in the vessel. Taking into account the residual retentions, it seems at present that the underwater dismantling scenario can be applied.

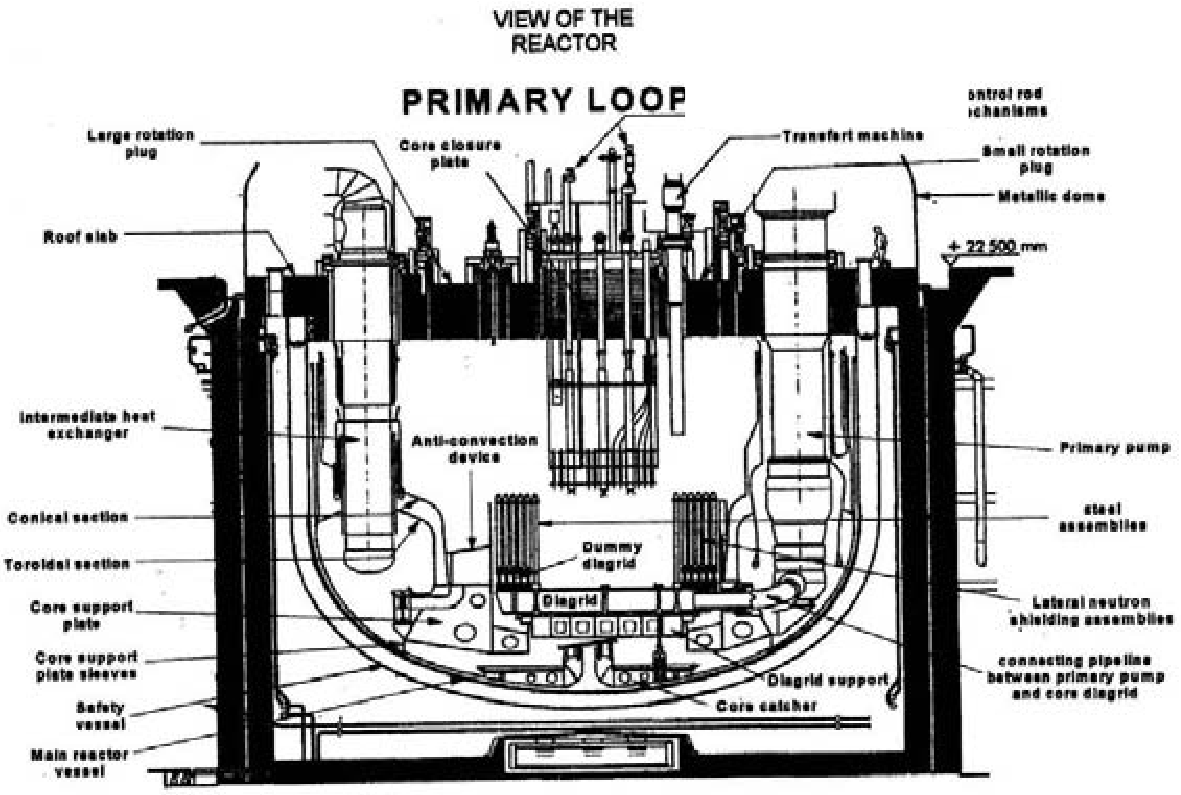


FIG. 1. View of the reactor.

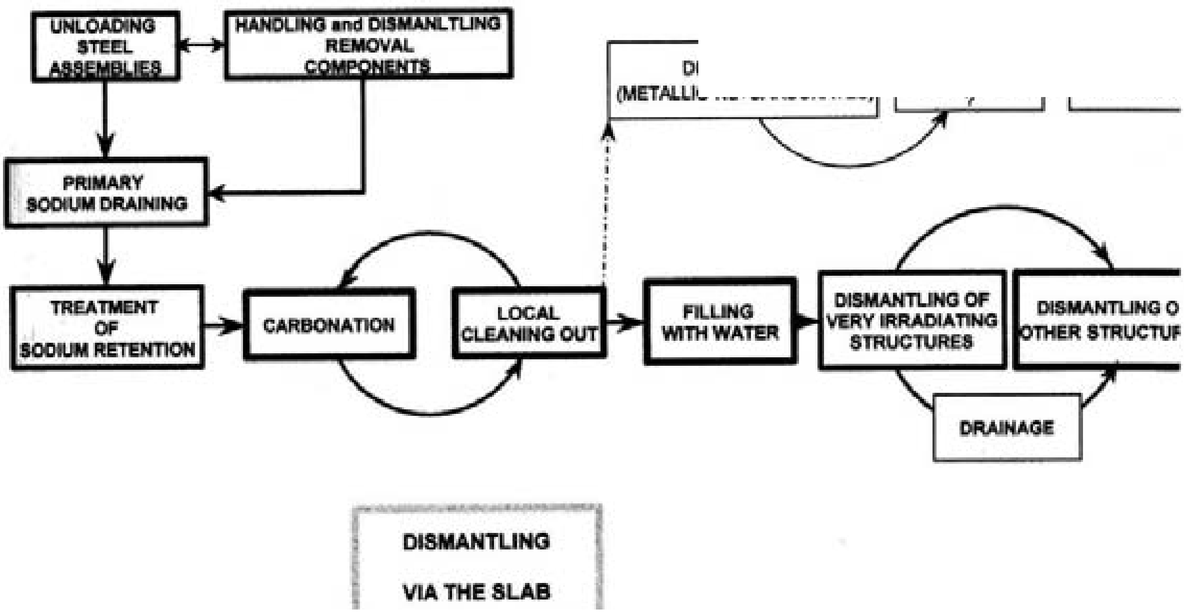


FIG. 2.Dismantling methods envisaged.

DECOMMISSIONING OF THE RAPSODIE FAST REACTOR: DEVELOPING A STRATEGY

J-M. GOUBOT, J. FONTAINE
COMMISSARIAT A L'ENERGIE ATOMIQUE (CEA), MARCOULE,
BAGNOLS-SUR-CEZE, FRANCE

X. BERSON, M. SOUCILLE
COMMISSARIAT A L'ENERGIE ATOMIQUE (CEA), CADARACHE,
SAINT PAUL LEZ DURANCE, FRANCE

Abstract

The RAPSODIE experimental fast neutron reactor at Cadarache (France) was operated from 1962 to 1982. The initial decommissioning operations began immediately, reaching IAEA stage 2 in 1994. Since then, the facility has been maintained under surveillance pending final dismantling scheduled to begin in 2020. New studies are now in progress to accelerate the dismantling process. The present status of the reactor block is described, the advantages and drawbacks of early dismantling are considered, and various dismantling scenarios are discussed.

1. INTRODUCTION

RAPSODIE is an experimental fast-neutron reactor with two primary sodium-sodium loops and two secondary sodium-air loops. Its initial power rating of 24 MW•th was subsequently increased to 40 MW•th. It was built and operated to design and qualify fuel elements for fast breeder reactors.

RAPSODIE was shut down in 1982 following leakage of sodium and nitrogen aerosols in the double containment vessel. Final shutdown operations began soon thereafter, all the core assemblies were withdrawn, cleaned and stored in the pool from 1984 to 1986. The primary sodium was drained into a tank and purified of ^{137}Cs. The purification campaign consisted in passing the liquid sodium (37 tons) through cesium traps.

Decommissioning to the safe enclosure stage was carried out between 1987 and 1994: 468 reflector assemblies (222 made of nickel, 246 of steel) were removed and washed to eliminate the residual sodium. The measuring instruments and control rod mechanism were removed from the core and packaged in dedicated containers.

After cutting the pipes, the primary vessel was isolated. The sodium was then eliminated from the primary circuits by in situ treatment with ethylcarbitol. A ^{137}Cs activity of 4.2×10^{11} Bq was removed from the surfaces. The circuits were decontaminated in three steps: the labile cesium was first removed by alkaline washing, followed by acid decontamination with Ce(IV) (removing 10% of the fixed contamination) and a final phosphatation step. The estimated contamination level was reduced from 5500 Bq/cm^2 to less than 10 Bq/cm^2. Dismantling could then proceed without constraints and the occupational dose was limited to an estimated 230 person·mSv.

The 37 tons of primary sodium were destroyed in the DESORA facility using the NOAH process (injection of small quantities of liquid sodium into a strong flow of aqueous sodium hydroxide). This operation took place from November 1993 to March 1994; 170 m^3 of 10 N soda were produced and sent to the *La Hague* plant [1].

An explosion occurred during cleaning of the residual sodium contained in the tank. After the accident [2], the main activity was to repair the damage with the objective of placing the facility under surveillance from 2005 to 2020 before final decommissioning.

Special financing has now been set up by the CEA to perform the dismantling operations. Final decommissioning of the Rapsodie reactor could be carried out at lower cost by shortening the safe waiting period. New orientations for the project can then be considered. Detailed studies were undertaken to submit a safety report to the Nuclear Safety Authority in 2006.

2. PRESENT STATUS OF THE RAPSODIE REACTOR

The reactor (Fig. 1) is inerted with nitrogen and sealed by a cover. The amount of residual sodium in the primary vessel is estimated 100 kg of sodium aerosols and oxides in the gas blanket zone, and 80 kg of metallic sodium, including approximately 20 kg on surfaces of structures and 60 kg in retentions. This was confirmed by borescope examination of the primary vessel (Figs 2 and 3).

The dose rate was measured in October 1989 and the maximum values (approximately 2×10^3 Gy/h) were found near the diagrids. Table 1 indicates the calculated mass and activity of the main reactor components as of January 2010.

The maximum activation (up to 10^9 Bq/g) is found on the stellite coated bushings located just beneath the core.

TABLE 1. CONDITIONS: JANUARY 1, 2010

Component	Mass, mg	Activation, TBq
Neutron shields	16.84	6.766
Thermal shield	5.18	0.315
Vessels	9.45	0.429
Diagrids	2.34	0.361
Core cover plate	0.68	0.732
Stellite-coated plates and bushings	0.05	28.439
Rare earth-doped concrete	166.71	40.104
Total	201.25	77.146

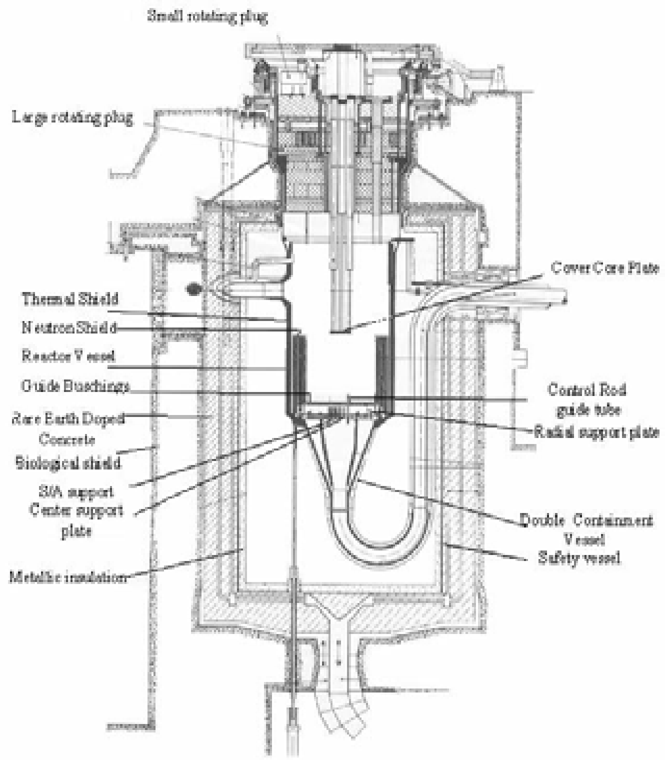


FIG. 1. Cross section of the Rapsodie reactor.

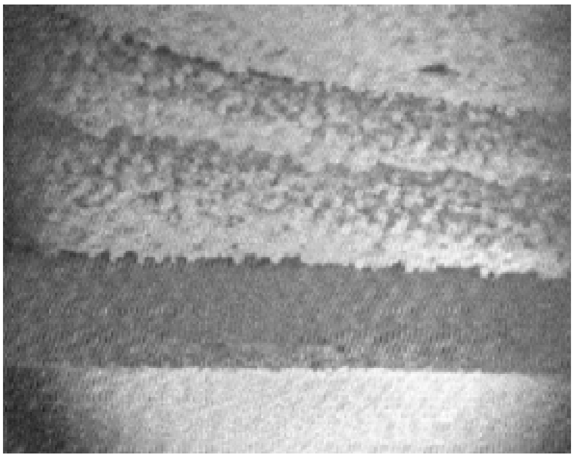


FIG. 2. *Aerosols under the rotating plate.*

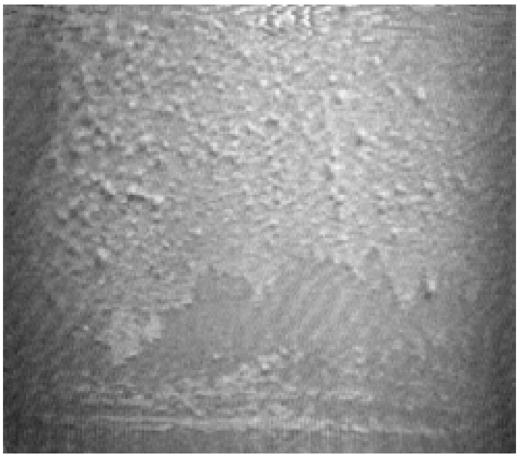


FIG. 3. *Sodium film on vessel wall.*

3. STRATEGY

Based on the results of international experience, it was decided to process the residual sodium from the outset of the operations. This approach rapidly excludes any sodium hazard, leaving a more conventional decommissioning situation for the remainder of the facility. This option also assumes that the sodium can be entirely processed, and may require specific development work (see § IV).

Remote dismantling is necessary because of the level of activation of the reactor structures (Table 1). The following diagram (Fig. 4) shows that three activation zones can be identified, corresponding to particular structural elements and suitable dismantling techniques.

Zone 1: corresponds to diagrid and neutron shielding elements, for which it might be advisable to partially fill the reactor with water to benefit from the strong radiation attenuation both for cutting devices placed in the reactor and for dimensioning the biological shielding.

Zone 2: corresponds to the thermal shields, reactor vessel, double containment vessel, safety vessel and part of the concrete radiological shielding. These items must also be dismantled remotely, but in air.

Zone 3: corresponds to the rest of the concrete radiological shielding, which may be dismantled by operators on contact.

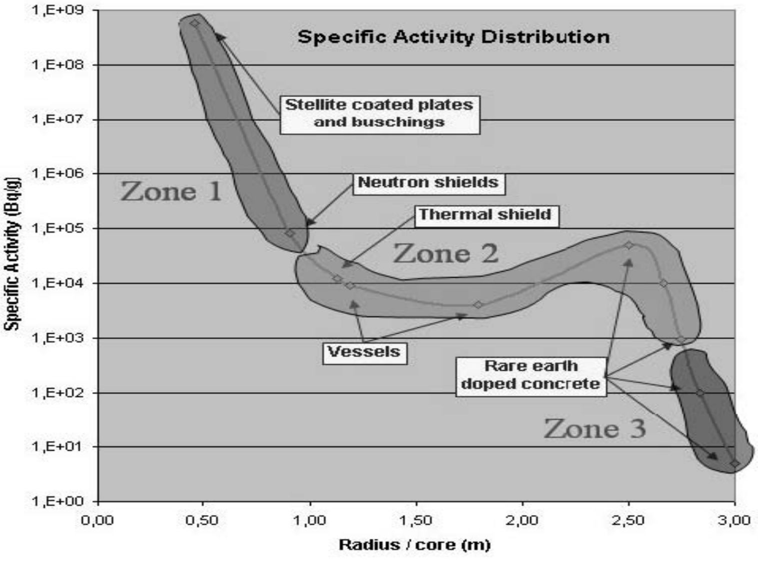


FIG. 4. *Specific activity distribution in the reactor core.*

Various types of dismantling scenarios can be proposed following the initial sodium processing step.

Installing a shielded enclosure to allow dismantling and waste packaging above the reactor would make the operations self-contained within the reactor block and would not interfere with the final decommissioning operations in the other buildings of the facility. This option is detailed as scenario 1.

An alternative would be to cut up subassemblies and transfer them for final cutting and packaging in an existing facility to limit the investment costs. For reasons partly related to the difficulties of developing transport casks suitable for large irradiating parts, it was decided not to use a facility outside RAPSODIE. Using the existing RAPSODIE dismantling cell was considered, but this option had few advantages compared with scenario 1.

Another option would be to form a single waste package consisting of the reactor vessel and all the internals. This alternative is based on the possibility of shipping large packages to the disposal site. In this case, the objective is to simplify the operations necessary on the sockets and diagrid components. This option constitutes scenario 2.

4. SODIUM PROCESSING

Sodium neutralization involves two steps. Generalized carbonation will eliminate sodium aerosol deposits, films, and a fraction of the thick retentions; the residual sodium inaccessible to carbonation will then be eliminated by local treatment. The sodium hold-up in the reactor is currently estimated as follows:

- 50 kg in the periphery of the diagrid;
- 5 kg on the support plate of the small diagrid;
- 6 kg retained at the bottom of the inlet pipe;
- 4 kg at the bottom of the tapered section of the inlet pipe (thermal shield vat).

4.1. Carbonation process

The carbonation process [3] consists in sweeping the vessel above the sodium by an inert gas (nitrogen) carrying carbon dioxide and a small quantity of water. The moisture in the gas is maintained below the room-temperature saturation limit to prevent condensation on the vessel walls.

The sodium (Na) thus reacts slowly to contact with moisture, and is converted to sodium hydroxide (NaOH); a limited flow of hydrogen is produced and removed by the scavenging gas. The sodium carbonates are inert, stable products, even in air, and are easily diluted in water. The carbonation process proceeds at a very slow rate. The treatment requires several months to be both effective and compliant with the safety criteria imposed on this particular operation. Under optimum conditions the carbonation process is capable of treating a sodium thickness of 50 mm in less than 1500 hours. The rate (0.8 mm/day) is practically constant throughout the process.

4.2. Sodium retentions

Two types of retention that will be not fully dealt with by carbonation have been identified in the reactor.

A massive 50 kg sodium metal retention around the diagrid (Fig. 5), which will be only partially treated by carbonation, as the time necessary for complete elimination was considered too long. A direct process for massive sodium deposits will be applied. The free surface area of this sodium retention is large, which is favorable to the process, although the equipment accessibility is limited.

Possible methods include direct application of water on the retentions, or mechanical treatment.

Tailoring specific techniques to the particular features (quantity, accessibility, etc.) of each retention point is now the subject of development in CEA test facilities at Cadarache. After neutralization of the sodium, the carbonates will be rinsed and the reactor vessel will be filled with water to eliminate any trace sodium for subsequent waste management.

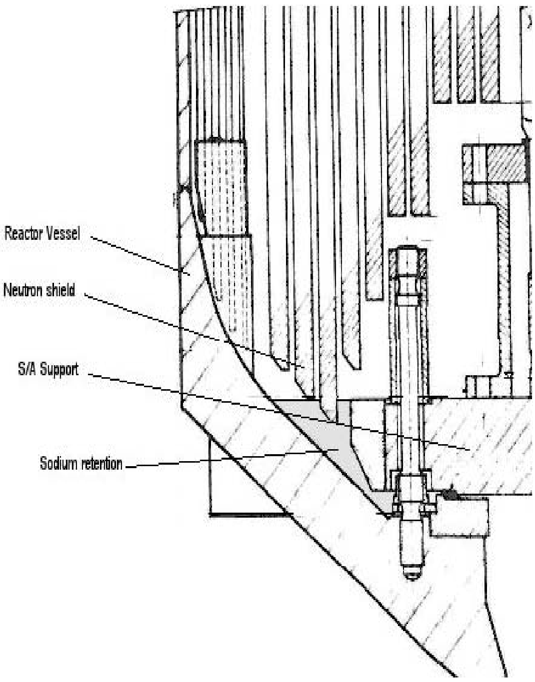


FIG. 5. Large sodium retention.

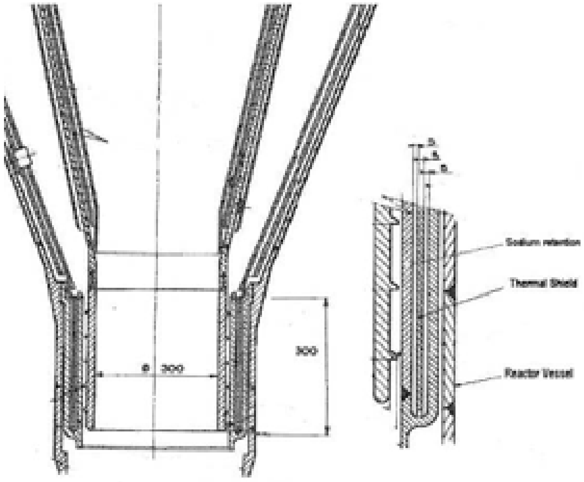


FIG. 6. Small sodium retention.

5. DISMANTLING SCENARIO n°1

5.1. Shielded enclosure

In order to preserve the reactor containment, a shielded enclosure will be built on the upper slab. As the water-filled vessel will provide a degree of biological protection, the shielded enclosure will include steel walls only 20 cm thick weighing 370 tons. The thickness is determined by the activity of the first neutron shield, which is the most irradiating part of the internals to be dismantled in the enclosure. The shielded enclosure (Fig. 7) will include:

- Working cell (with a multipurpose extended-reach manipulator, a 15-ton crane, hoisting gear and manipulators) located just above the vessel,
- Handling cell (handling robot, cutting tools),
- Maintenance cell,
- Measurement area where the wastes are characterized,
- Conditioning cell where the wastes are packaged and transferred outside. The conditioning and packaging cell can accommodate 5 m^3 and 10 m^3 containers.

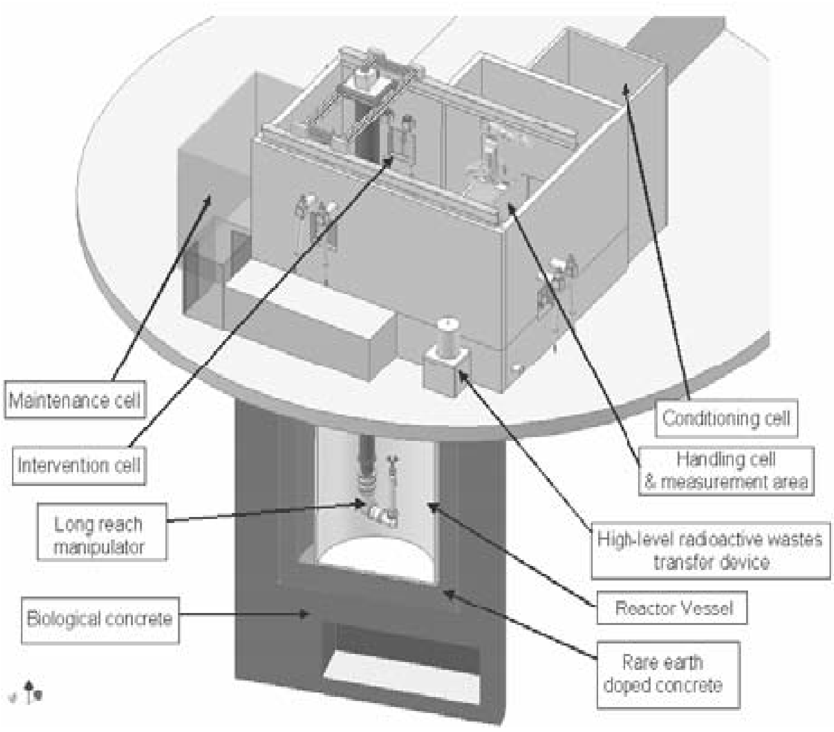


FIG. 7. Shielded enclosure.

5.2. Reactor disassembly

As the shielded enclosure is set up on the reactor, the rotating plugs can be removed to gain access to the reactor internals. The rotating plugs consist of a control plug, a small plug and a large one (Fig. 8). They will be removed manually. The layers of borated graphite will be removed and placed in specific containers. The bottom part of the control plug will be cut (Fig. 1) but not removed immediately, as the core cover plate is too radioactive for direct operation. It will remain in the vessel underwater.

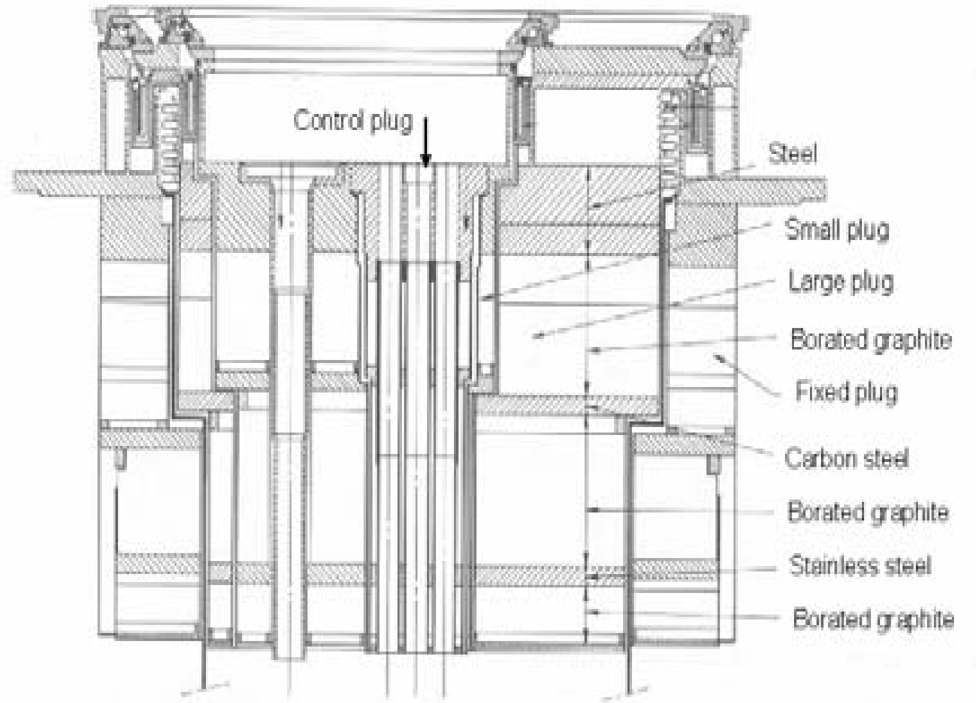


FIG. 8. Rotating plugs.

The reactor will be dismantled from the inside out. The neutron shields comprise ten concentric steel cylinders. Each cylinder will be cut azimuthally using an underwater plasma torch, mounted on a self-positioning platform (by means of special jacks). The rings will then be lifted into the enclosure and cut up into smaller pieces in the handling cell. Depending on the radioactivity of the internals, concrete shielding up to 40 cm thick may be used in the containers. The bottom part of the control plug with the cover core plate will then be removed.

The final underwater operation will involve disassembly of the diagrids and bushings. The radioactivity levels of these parts are too high to allow conditioning in the enclosure. To evacuate these materials, a shielded container will be moved into the vessel and filled with high-level radioactive waste under water. The container will then be raised into the cell and transferred to a dedicated storage facility at Cadarache. The vessel will then be drained and subsequent operations performed in air.

Large parts of the thermal shields and the double containment will be cut using a plasma torch in the same way as the neutron shields. The vessel and the ‘saxo’ pipe will be dismantled with a cutting tool carried by an extended-reach manipulator.

The safety vessel consists of large plates and a multilayer thermal shield. A large basket will be lowered from the enclosure, and the plates will be cut and laid in the basket before hoisting it back into the conditioning cell.

5.3. Demolition of the concrete

The Rapsodie reactor is surrounded by a specific grade of concrete to ensure neutron and gamma protection. The material (known as Sercoter concrete) is doped with serpentine, corundum and rare earth elements. The main radioisotopes found in the concrete at this time are ^{152}Eu and ^{154}Eu. The concrete mass also includes a large number of cooling pipes.

To dismantle this material, the extended-reach manipulator will be equipped with a pneumatic jackhammer. When the pipes are accessible, they will be cut out and removed. A remote-controlled demolition machine will be set up at the bottom of the vessel and used to load the concrete rubble and metal pipe sections into the baskets.

The computational results show that the lower part of the cavity is not very radioactive, suggesting that the final dismantling of the Sercoter concrete could be carried out manually.

5.4. Waste management

The balance of the radioactive waste containers shows that only 40 high-activity containers (70 L each) will be necessary to dismantle the reactor (Table 2).

TABLE 2. Waste balance: January 1, 2010

Type of container Number High-activity shielded containers (70 40 L) 5 m^3 container – 40 cm concrete 28 shield 5 m^3 container – 30 cm concrete 26 shield 5 m^3 container – 20 cm concrete 12 shield 5 m^3 container – 10 cm concrete 3 shield Standard 5 m^3 container 55 Graphite container (10 m^3) 5 Very low-activity waste container 8 (2 m^3) Very low-activity waste container 39 (1 m^3)

The shielded containers are used for stellite-coated waste. No final repository is currently available in France for these containers, which will be transferred to an interim storage facility at Cadarache.

The graphite dose rate is expected to be low but the presence of long lived radioisotopes currently precludes disposal in France. It will be probably transferred to an interim storage facility. The other containers could be shipped to the ANDRA storage facilities for low-level and very-low-level waste.

6. DISMANTLING SCENARIO n°2

The purpose of this scenario is to provide a veritable alternative that minimizes the operations and equipment used for remote dismantling of the reactor block.

The complete sequence of disassembly and cutting operations necessary to dismantle the vessel and internals according to scenario 1 would require about 1 year. The major difference in scenario 2 is that the vessel would be detached from the reactor block and constitute a single package.

6.1. Rotating plugs and new dismantling rotating plug

With the reactor vessel filled with water for operator protection, a light containment structure would be installed. The plugs would be dismantled directly by the operators. Additional shielding would be used for cutting the lower portion of the core cover plug and placing it on the diagrid.

On completion of dismantling, a new interface would be installed on the reactor. The new plug (Fig. 9) would be designed to allow the use of a teleoperated arm mounted at the tip of a various lengths mast.

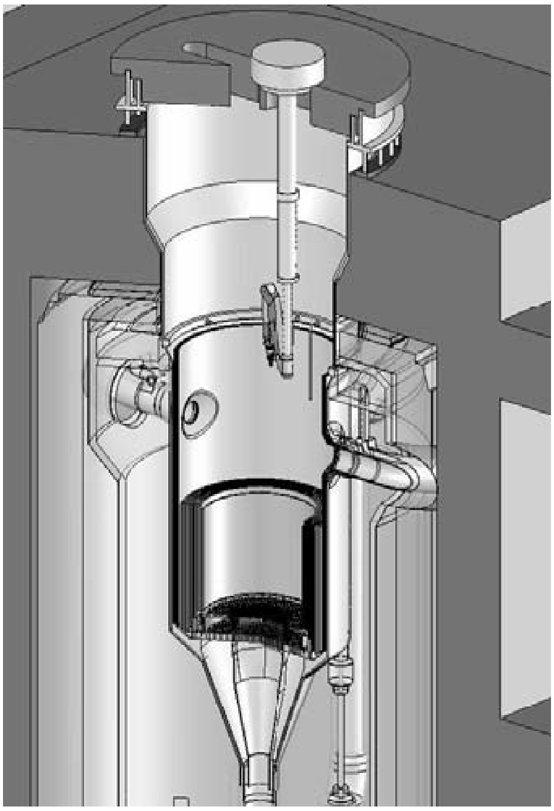


FIG. 9. New dismantling rotating plug.

6.2. Separation of the vessel and preparation of the package

The RAPSODIE safety vessel is suspended from the top of the reactor block and includes the following connections:

- 3 sodium inlet/outlet pipes;
- 5 small pipes at the top of the vessel;
- 4 connections for diagrid displacement measuring devices;
- Lateral reinforcing elements at the free sodium level.

The only reasonable access routes are from the top (inside) and from the bottom (outside the vessel). The vessel would gradually be detached and the cutup parts placed on the diagrid. The weight of the vessel would be supported from a flange before it is completely detached, and the same flange would be used to hoist the vessel into the transport cask (Fig. 10). Table 3 indicates the package characteristics. After injecting concrete grout, the cask would be removed from the reactor block on a trailer and sent to the disposal site.

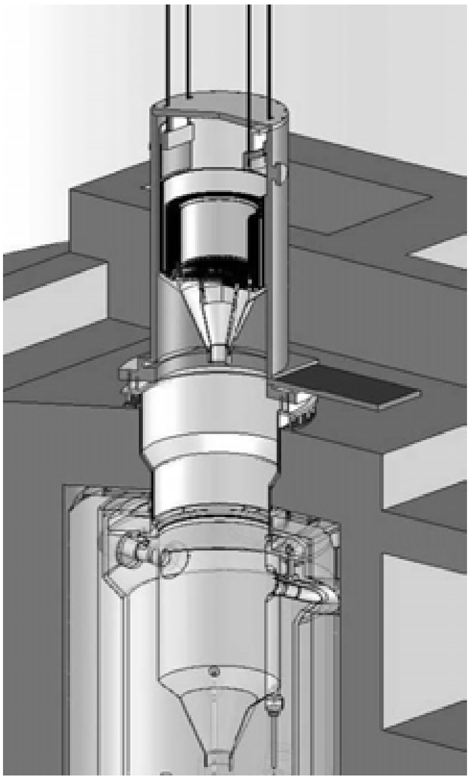


FIG. 10. Hoisting the vessel into the cask.

TABLE 3. PACKAGE CHARACTERISTICS

Property	Value
Vessel outside diameter	2.4 m
Height of removed vessel	5.4 m
Cask diameter	3.0 m
Cask height	6.5 m
Total package weight	142 tons
Weight of waste items	32 tons
Cask empty weight	80 tons
Weight of injected grout	30 tons

6.3. Final operations

As the most irradiating parts have been removed, a shielded enclosure including steel wall 8 cm thick weighting less 100 tons will be built on the upper slab.

The simplified cell will be used for containment, dismantling, waste packaging, and equipment maintenance operations (Fig. 11).

The manipulator arm used for cutting the vessel will be mounted on a mast with 3 degrees of freedom, secured at the top and bottom in the reactor. The remaining operations will be practically the same as for scenario 1.

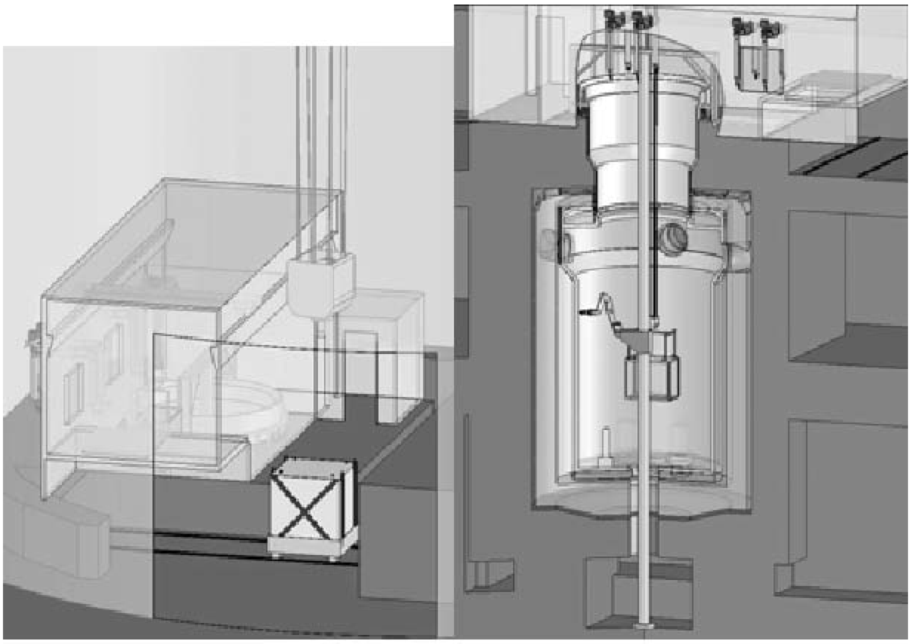


FIG. 11 Simplified dismantling cell.

6.4. Waste management

The package volume (cask + vessel) is about 46 m^3. In scenario 1.26 bins measuring 5 m^3 each would be necessary to contain the same waste materials, i.e. a total waste package volume of 130 m^3. The acceptability of the package by the disposal center appears to be feasible, but further investigation will be necessary in the following areas:

- Safety and technical options for disposal;
- Verification of the type of transport package required;
- Further assessment of impurities in the vessel material and of the corresponding long-lived radioisotopes.

7. CONCLUSION

Studies are nearing completion to determine the dismantling policy for the RAPSODIE reactor. They indicate that it would be advisable to resume decommissioning operations in the near future.

Two very different scenarios have been proposed and are now being comparatively evaluated.

Scenario 1 uses proven technical solutions (a shielded enclosure has already been used for dismantling purposes [4]).

Scenario 2 provides an interesting solution to limit cutting operations inside the reactor, but several aspects of preparation and transportation of the waste package must be examined in greater detail.

A request for a dismantling license will be submitted to the safety authorities in 2006. If the data required to make a decision are not available by that time, both scenarios could be submitted for evaluation.

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POTENTIAL IMPACT OF SODIUM FAST REACTOR (SFR) DECOMMISSIONING EXPERIENCE ON FUTURE GENERATION IV SFRS DESIGN

G. RODRIGUEZ, E. GOUHIER
COMMISSARIAT A L'ENERGIE ATOMIQUE (CEA), CADARACHE,
SAINT PAUL LEZ DURANCE, FRANCE

Abstract

In comparison with some other GENERATION IV (GEN IV) reactor concepts, Sodium Fast Reactor (SFR) concept has the main advantage that prototypes of this system have been built and successfully operated in many countries. Some of these prototypes have been shut down and are presently being decommissioned. The importance of decommissioning this type of Reactor cannot be underestimated, in support of the SFR as a viable reactor concept for the future, for the following reasons: To provide guidelines on how future reactors should be designed to optimize the performances in terms of operational behaviour, inspection, maintenance & repair; To provide information on how future reactors should be designed to reduce decommissioning costs. To inform about the performance of construction's materials that have been exposed to sodium, radiations, high temperatures and temperatures cycling; To develop new technologies that will facilitate the implementation of improved operational characteristics of future SFRs. Thus even if decommissioning has to be considered as a long term study, significant cost savings in the decommissioning of future systems can be identified from this actual experimental feedback. There are now some innovations being designed from the decommissioning process in the fields of decontamination processes, robotics and waste management which will bode well for the successful operation, maintenance and inspections of future SFRs. The objective of this paper is to give an overview of SFR decommissioning experiences. It is highlighting what are major issues that would need to be addressed in future basic SFR design concept to facilitate the decommissioning. This experience covers all the different basic systems defined in a 'typical' Sodium Fast Reactor design (i.e. primary circuit components, secondary circuit components, cold traps, steam generators, sodium tanks, etc.). The recommendations proposed will have the following objectives: Be easy to proceed and implement; Will not interfere with the efficiency of the reactor; Will not lessen the inherent safety of the reactor; Will reduce future decommissioning costs. Some estimated costs of SFR decommissioning operation are given from the experimental feedback of the actual decommissioning operations.

1. INTRODUCTION

In the main goals to be achieved in the GENERATION IV future reactors, one is relative to the economic competitiveness of these reactors. The objective is to obtain construction cost equivalent to the most worldwide reactor design built all over the world: Pressurized Water Reactor. If construction cost is a task to achieve, it must not be forgotten that decommissioning cost is also an intrinsic part of the price and in that specific field, GENERATION IV reactor should also be as competitive as other commercial reactors. The specific history of Sodium Fast Reactors development has lead to a situation where now this design has been selected for GENERATION IV program, and at the same time major SFR already build are now under decommission, mainly in the western countries. Therefore it is possible to evaluate right now with real example and not only with global estimations, if the decommissioning costs of past SFR is of the same order than actual PWR. Relatively to the previous conclusions, it will be possible to give certain recommendations on the future GENERATION IV (GEN IV) design to reduce the future decommissioning cost without modifying (or with a minimum influence) the target performances. This study will be made taking into consideration mainly the following reactors:

- Superphénix (France): demonstration reactor (3000 MW•th/1250 MW(e)) under decommissioning since 1997 after a premature final shutdown for political decision, and after only 12 years operations;
- Phénix (France): prototype reactor (560 MW•th/250 MW(e)) in operation until 2009 and will reach at that time 35 years operation. Nonetheless an opportunity study has been made in 2003 to make an early estimation of the decommissioning costs of Phénix taking into account the experimental feedback gained by the different SFR already under decommissioning phase.

Of course, this study will also refer to some other SFR decommissioning operations as RAPSODIE (France), EBR II, Fermi and FFTF (USA), KNK-II (Germany), PFR and DFR (UK), BN350 (Kazakhstan). Plan for this study will be the following:

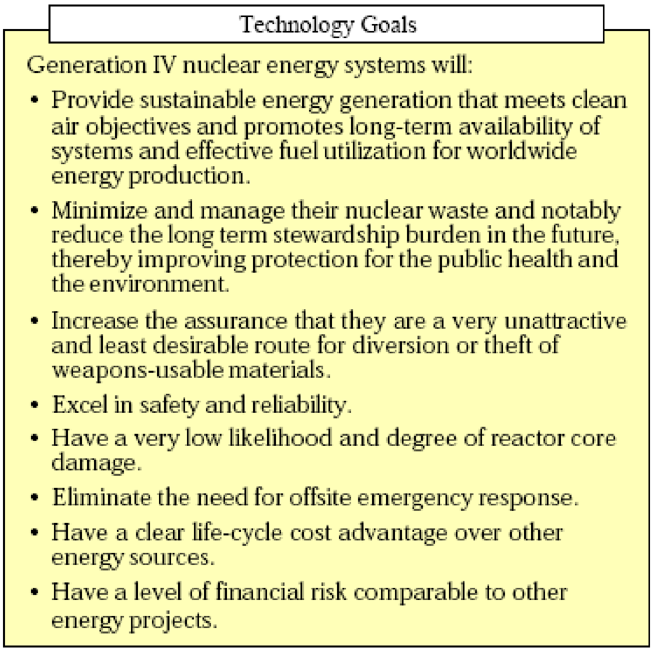
- A general survey of decommissioning SFR estimation costs compared to other reactors will be made. It would allow estimating where the main differences are and where gains can be achieved;
- A general description of the SFRs decommissioning strategy will be made;
- It will be tried to identify different sensitive points that significantly affect costs during the plant decommissioning: primary circuit, primary components, secondary circuits, secondary components, steam generator, etc;
- A list of recommendations from decommissioning actual and future SFRs will conclude this study.

2. DECOMMISSIONING A NUCLEAR REACTOR: MAIN GOALS TO ACHIEVE

Decommissioning operations can be divided in several basic operations. The main objective of decommissioning a nuclear reactor is to accomplish its dismantling in respect with safety, minimizing collective dose rate of operators, minimizing radioactive wastes production, in a restricted time and with the constant idea to minimize or reduce costs. All these operations must be conducted according to administrative authorization and safety agreements. Even if decommissioning can be considered as a long time overall project (from 10 to 30 years), the objective to control decommissioning duration remains crucial. Indeed, every year spent means one full year to pay for reactor maintenance and monitoring. Therefore in decommissioning options, delays can be associated lost of money.

In France, the French electricity supplier EDF, is providing funds for the future decommissioning of the current reactors – as well as for future wastes and used fuel management disposal. This fund is defined as a tax included in the electricity price. Decommissioning price is estimated up to now to 10-15% the reactor capital cost. Fuel management from mining to waste disposal, reprocessing and transport is estimated to around 20% of the total capital cost [1].

These values are based on estimations and are mainly settled for commercialized Pressurized Water Reactor (PWR) type. Therefore the objective of GEN IV reactor is to be economically competitive to PWR even in the decommissioning costs. Indeed, amongst the technology goals GEN IV reactors has to fulfill, the two last have direct or indirect links with decommissioning tasks (see Fig. 1).



Technology Goals

Generation IV nuclear energy systems will:

- Provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.
- Minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment.
- Increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials.
- Excel in safety and reliability.
- Have a very low likelihood and degree of reactor core damage.
- Eliminate the need for offsite emergency response.
- Have a clear life-cycle cost advantage over other energy sources.
- Have a level of financial risk comparable to other energy projects.

FIG. 1. Technology goals to achieve for GENERATION IV reactors.

Decommissioning costs decomposition can be done in several ways. One usually used is the following:

- Item 1: Costs due to technical decommissioning operations;
- Item 2: Costs due to used fuel final treatment;
- Item 3: Costs due to the final disposal of radioactive wastes produced (from high level wastes until very low level wastes);
- Item 4: Costs due to reactor maintenance and surveillance during all the decommissioning operations. This task is called SENEX in France meaning (Surveillance, Entretien (*Maintenance)* and Exploitation: Long term fixed costs). The acronym SENEX will be kept in the following text.

To optimize costs in all these items, it is necessary to:

- Design future reactors in a way that they will be technologically easily decommissioned (impact on item 1, indirectly on item 4 by reducing time);
- Select fuel suitable for final treatment as wastes or for reprocessing (impact on item 2);
- Design concepts and materials that will minimize the volumetric quantity of wastes and favor the lowest quantity of radioactivity and minimize long time radionuclides (impact on item 3);
- Prepare in advance the decommissioning program to avoid delays during the decommissioning operations due to unexpected technical difficulties (impact directly on items 1 and 4).

3. WHAT ARE THE MAJOR DIFFERENCES BETWEEN DECOMMISSIONING A SFR AND A PWR?

The difference of coolant between PWR and SFR is one of the major differences between these two concepts. In PWR, water is not considered as a hazardous chemical product. It can be drained, transferred, stored and purified with no major chemical hazards, except its own contamination. On the opposite, sodium is considered in itself as a hazardous chemical product and will be handled with cautions as long as it exists in its metallic form. Therefore the technical decommissioning operations are more complex as far as there is still metallic sodium inside the plant. Moreover, sodium technology requires a specific technology and know-how to handle it. Technological designs required to its safe management are concepts compared to water coolant: nitrogen or argon cover gas, pre-heating piping and vessels, smokes and fire detectors, sodium leak detectors, specific In Service Inspection component.

Hence, the safe conversion of metallic sodium to stable chemical product for final depository will induce several specific processes (involving several treatment facilities). Sodium treatment processes are today under industrial qualification and development. This experimental feedback will be suitable for future SFR decommissioning program and only minor R&D adjustment will be necessary. Intuitively, it can be thought that decommissioning a SFR will be more expensive than a PWR because of the additional sodium waste treatment and handling. Moreover almost 50% of the nuclear electricity produced in the world is coming from PWRs. In France, except Phénix, all the nuclear electricity is today produced by PWRs. Therefore for PWR, the reproducibility factor will allow a significant gain in experimental feedback, improvement, common decommissioning strategy and well trained specialized industry. For SFRs there is at the present time not even two similar design reactors in the world. In Europe the first SFR standard design should have been the EFR project, but it has been achieved and abandoned without realization in 1998.

4. IS DECOMMISSIONING A SFR MORE EXPENSIVE THAN A PWR ?

4.1. Nuclear reactor decommissioning costs estimates

NEA has recently carried out a study on decommissioning policies, strategies and costs [2, 3]. Table 1 is extracted from [2].

Roughly proposed the following cost weights by subject are shown in Table 2.

TABLE 1. DECOMMISSIONING COST ESTIMATE

Reactor type (no of data sets)	Average cost €/kWe (US$/kWe)	Standard deviation €/kWe (US$/kWe)
PWR (21)	260 (320)	158 (195)
VVER (8)	268 (330)	89 (110)
CANDU (7)	292 (360)	57 (70)
BWR (9)	341 (420)	81 (100)
GCR (4)	> 2029 (> 2500)	-

Legend:	
	PWR: Pressurized Water Reactor;
	VVER: Russian type of Light Water Reactor;
	CANDU: Canadian type Heavy Water Moderated Reactor;
	BWR: Boiling Water Reactor;
	GC: Gas Cooled reactor (first generation design, Natural Uranium Graphite Gas type).

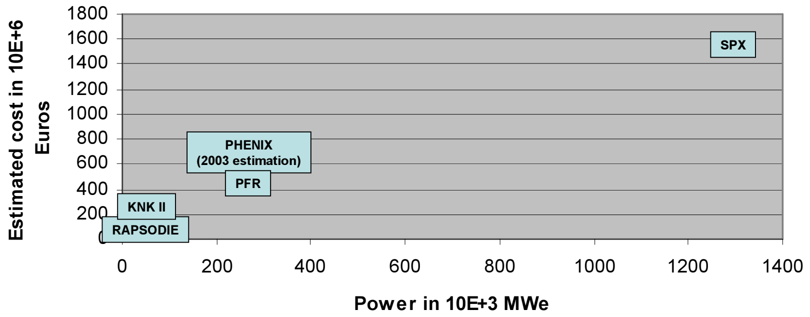
TABLE 2. COST REPARTITION BY ITEM

Item	Estimated % of total decommissioning cost (NEA estimation)	Estimated % of total decommissioning cost (CEA estimation)
Dismantling (Item 1 in the present classification)	25 to 30%	25%
Waste management /disposal (Item 2 and 3)	25 to 30%	30%
Security, survey and maintenance (Item 4)	10%	30%
Site clean-up and landscaping	10%	Included in other tasks
Project management, engineering and site support	10%	15%
Other tasks	<5%	X

4.2. SFR decommissioning cost estimates

With the early design and construction of Sodium Fast reactor in nuclear generation and also with their early stop for some of them, there is a valuable experimental feedback on decommissioning SFRs. Therefore some relative costs can already provide information (and not only estimations) about the price of dismantling operations. Recent study made to prepare the future decommissioning of Phénix reactor (planned for 2009) has compiled estimations of the different decommissioning costs already announced for French reactors (SFR) and Worldwide (SFR). These values are often difficult to compare because in the decommissioning costs some are including SENEX costs, some other projects are not and are only considering Item 1, 2 and 3 in cost evaluation. Therefore values presented in Fig. 2 must be considered with caution. Some of them are only estimation with 30% accuracy. The values must be compared as a general trend and not for their exact value.

These figures must be considered as global estimation, and an effort must be done to provide cost decomposition, to compare them with the other costs given for example in Table 1 from NEA or from other data presenting more accurate decomposition costs. Neither the less these estimated cost remains greater than those presented by NEA for other reactor type, especially compared with PWRs.



Legend: *KNK-I: German experimental sodium fast reactor,*
RAPSODIE: French experimental sodium fast reactor, PHENIX: French prototype sodium fast reactor,
PF: British prototype sodium fast reactor,
SPX: French Demonstration sodium fast reactor

Note: For KNK-II and RAPSODIE it has been assumed a potential electrical yield of 0.4 because these reactors were producing only thermal power

FIG. 2. Comparison of the estimated decommissioning cost of several nuclear reactors [5].

5. WHERE GAINS COULD BE MADE IN SFRS TO ACHIEVE COMPETITIVE DECOMMISSIONING COSTS?

The experimental feedback from the decommissioning of present Fast Reactor is an incomparable asset to minimize future R&D development for sodium treatment. The following recommendations will allow reducing SFR decommissioning costs:

- Design compact reactors: decommissioning price is directly linked to the volume of wastes generated. Designing compact reactors will reduce this cost.
- For decommissioning, time is playing and ambiguous role. With time, radioactivity is decreasing and the ALARA (As Low As Reasonably Achievable) principle should recommend waiting for a while for radioactive decrease at least for short time radionuclides. This would go in a way of radioactive waste declassification and collective dose rate reduction and simplification. On the other hand, waiting means loosing knowledge, loss of team motivation, increase of costs due to SENEX, confrontation with stronger safety rules and more expensive waste issues. In 2000, EDF has decided to apply level 3 (back to the grass) without delay between level 2 and level 3, for the decommissioning of all their old reactors (Generation 1 plus SPX). The French National Safety Authority has accepted this strategy [4]. This change in the company strategy is showing that economically, waiting is not a unique issue. The Rapsodie strategy has also changed by starting the dismantling now instead of waiting until 2020.
- So today, the tendency seems to be: ‘the quicker, the better’. Principally for SFRs, SENEX costs should be significantly reduced when metallic sodium will have totally disappeared of the plant area. Safety surveillance could be then reduced. This means that sodium treatment must be done in the shortest time that is conceivable. To success in this principle, it means that the operating works must be prepared in advance and having assurance that waste issues for liquid, gas releases and solid materials will not block the overall process.
- SENEX reduction cost is one of the major objectives. As an example the SENEX cost is estimated to represent around 40% of the total decommissioning cost of PHENIX reactor.
- SENEX can be consequently mastered and reduced if the following strategy is respected :
  (i) An adaptation of personal to future decommissioning works (training to be made in advance).
  (ii) An accurate optimization of liquid effluents production and issue.
  (iii) A minimization of the equipment to maintain, principally those required for safety during reactor operation. A general study of the future need of the plant in the future and existing old equipments is useful. Often, money can be saved by getting rid of the existing old and over dimensioned equipments and by replacing them by new equipments fit for future requirements of the reactor in a decommissioning perspective.

6. IS THERE A COMMON STRATEGY TO DECOMMISSION SFRS ?

An overview of all the SFRs under decommissioning is showing several strategies and technical issues selected. The solutions are strongly linked to conditions inherent to the reactor design and to national context (legislation, long term strategy of respective nuclear industry or countries). Moreover except SUPERPHENIX, all the SFRs under decommissioning are prototype or experimental reactor. Therefore there was no real harmonization between the different national SFR designs. This lack of harmonization implies several technical decommissioning solutions are specific to one reactor. As examples:

- Dounreay Fast Reactor is the only SFR operating with eutectic NaK coolant. This specific coolant is modifying the decommissioning strategy.
- RAPSODIE, KNK-II and SUPERPHENIX have decided to unload all their assemblies without dummy core. PFR defuelling was done with a dummy core.
- After defuelling, primary sodium was drained in one big vessel storage for RAPSODIE, and in multiples 200 L drums for KNK-II. For PFR and SPX, the strategy was to keep primary sodium liquid in the reactor vessel until its final treatment.
- The issue of the primary sodium treatment is made with the NOAH process for RAPSODIE, PFR and SPX (planned for Phénix). For EBRII, Fermi, FFTF and BN-350 another process has been chosen.

Nevertheless it is possible to define a general trend on decommissioning SFRs strategy. This trend is moreover more precise if the following reactors are compared: PFR, SPX and Phénix future project. They have some similarities in their respective design, and a common approach for sodium treatment has been chosen with the NOAH process [6]. The general strategy is the following [7]:

- Keep primary sodium inside the reactor vessel until its final online treatment. This choice is justified because primary vessel answers to all safety requirements to store active sodium. And no tank was available to store this large amount of sodium. The design and construction of storage tanks would have implied costs and more metallic low level radioactive wastes.
- Keep sodium in liquid state with additional heating devices adapted on the primary vessel.
- First operation consists in unloading the fuel and fertile elements plus control rods to remove radioactive material risk. In case of Phénix and SPX steel assemblies are also withdrawn.
- In parallel, secondary loops are drained, and isolated from primary circuits. They can be early decommissioned.
- Primary sodium is online treated with NOAH process producing concentrated sodium hydroxide liquor (10 mol/L). This liquid effluents is then neutralized and released to the sea (PFR) or transformed into concrete blocks for final disposal (SPX).
- On PFR and SPX, technical operation must be envisaged to allow better sodium draining of primary vessel. These techniques are mainly: whole drilling, local siphoning.
- Residual amount of sodium is then treated using divided water carried in gas vector. In the UK gas vector is nitrogen and the process is known as Water Vapor Nitrogen process. In France gas vector is a mixture of nitrogen and carbon dioxide and the process is known as carbonation process [8].
- After this treatment, the final decommissioning of the vessel is envisaged under water for SPX.
- Removable components and cold traps will be treated with adapted processes.

A general trend can be drawn by comparing the decommissioning strategy of PFR and SPX reactors, and Phénix future project. Therefore, on the basis of this general trend, it is possible to analyze its experimental feedback and how it could influence new SFR GEN IV reactor design.

7. GENERAL TREND ON DECOMMISSIONING SFRS

Specific chemical properties of sodium coolant will impose the guideline of decommissioning operations. They are related to sodium specific treatment and can be defined as follows:

- Radiological inventory of components and coolant;
- Functional simplification of means and maintenance facilities;
- Maintaining sodium liquid with other means than primary pumps in operation;
- Secondary loops draining and isolated from primary circuits;
- Nuclear material unloading (fuel and breeder elements);
- Control rods unloading;
- Other assemblies unloading (steel and reflectors);
- Treatment of removable primary big components (primary pumps and intermediate heat exchanger);
- Treatment of removable primary small components;
- Treatment of removable secondary component;
- Primary sodium treatment;
- Secondary sodium treatment;
- Complementary primary sodium draining;
- Treatment of residual primary sodium remained inside the vessel;
- Treatment of residual sodium of secondary loops and their decommissioning;
- Steam generator treatment and decommissioning;
- Primary vessel decommissioning;
- Treatment of exotic Na components: cold trap, cesium trap, bubbler …

Table 3 shows how these macro-tasks were solved by several SFRs: RAPSODIE, KNK-II, PFR, SPX and PHENIX project.

TABLE 3. DECOMMISSIONING STRATEGIES OF SEVERAL SFRS

Macro – tasks	**RAPSODIE solution**	**KNK-II solution**	**PFR solution**	**SPX**	**Phénix (project)**
Radiological inventory of components and coolant	Done	Done	Done	Done	In progress
Functional simplification of means and maintenance facilities	Done	Done	Done	Done	Planned
Maintaining sodium liquid with other means than primary pumps	Residual power plus additional circuits	Residual power plus additional heating circuits	NaK specific circuit	Trace heating of the lower part of the safety vessel (one third)	Under study
Secondary loops draining	Done, sodium stored in 2 tanks	Done, sodium stored in drums	Done, sodium stored in tanks	Sodium stored in tanks	Sodium stored in tanks
Nuclear material unloading (fuel and breeder elements)	Done without dummy core	Done without dummy core	Done with the original dummy core	Done without dummy core	Planned without dummy core
Control rods unloading	Withdrawn before 1^{ary} vessel treatment	Withdrawn before 1^{ary} vessel treatment	Previously let in the vessel but finally withdrawn	Withdrawn before 1^{ary} vessel treatment	Withdrawn before 1^{ary} vessel treatment
Other assemblies unloading (Steel and reflectors)	Withdrawn before 1^{ary} vessel treatment	Withdrawn before 1^{ary} vessel treatment	Left inside the vessel until its final decommissioning	Withdrawn before 1^{ary} vessel treatment	Withdrawn before 1^{ary} vessel treatment

Primary sodium draining	Done in an adapted existing tank: RENa 302	Done in several drums	Not done, sodium is kept in 1^{ary} vessel	Not done, sodium is kept in 1^{ary} vessel	Not already determined
Treatment of removable primary big components (PP and IHX)	Done in the existing cleaning pits	Done in the existing cleaning pits	Left in the vessel. Done at the same time as 1^{ary} vessel treatment	Withdrawn and treated in a specific area mainly under air then in the cleaning pit	Planned to be withdrawn and cleaned in the existing cleaning pits
Treatment of removable primary small components	Done in the existing cleaning pits	Done in the existing cleaning pits	Left in the vessel. Done at the same time as 1ary vessel treatment	Withdrawn and treated in a specific area under air	Planned to be withdrawn and cleaned in the existing cleaning pits
Treatment of removable secondary component	Done in air	Done in air	Done in situ with secondary loop treatment	Withdrawn and treated in a specific area under air	Best strategy to be determined
Primary Sodium treatment	Done with NOAH process/DESORA facility (30 kg/h)	To be treated by UKAEA	Done with NOAH process/SDP facility (130 kg/h)	Done with NOAH process/TNa 1 and TNa 2 facility (6 t/d)	Done with NOAH process/Facility to be defined
Treatment of resulting liquid effluents	Sodium hydroxide was reused by La Hague reprocessing plant	Same treatment as sodium coming from PFR	Sodium hydroxide is decontaminated, neutralized and released to the sea	Direct cementation of sodium hydroxide into Very Low Level Wastes cement blocks	Best strategy to be determined
Secondary Sodium treatment	Will be done with PX secondary sodium treatment	To be treated by UKAEA	Done with NOAH process/SDP facility (130 kg/h)	Done with NOAH process/TNa 1 and TNa 2 facility (6 t/d)	Done with NOAH process/Facility to be defined
Complementary primary sodium draining	Done by complementary draining with specific tools	Not necessary, good overall draining	Done by drilling operations and complementary draining with specific tools	Done by drilling operations and complementary draining with specific tools	Under study, only minor complementary draining are envisaged
Treatment of residual primary sodium remained inside the vessel	Planned by carbonation process	Not done	Planned by Water Vapor Nitrogen (WVN) process	Thorough carbonation until almost complete sodium transformation	Thorough carbonation until almost complete sodium transformation
Treatment of residual sodium of secondary loops and their decommissioning	Done under air without preliminary sodium treatment	Done under air without preliminary sodium treatment	Done under air after WVN or carbonation preliminary treatment	The question of carbonation preliminary treatment is under discussion	Best strategy to be determined. Waiting for SPX feedback
Steam generator treatment and decommissioning	No SG on RAPSODIE	No SG on KNK-II	Done under air after WVN preliminary treatment	Done under air after carbonation preliminary treatment	Done under air after WVN preliminary treatment
Primary vessel decommissioning	Done by underwater cutting tools	Done by under inert gas cutting tools	Done by under air cutting tools	Done by underwater cutting tools	Best strategy to be determined. Waiting for SPX feedback

Connection between level 2 and level 3 decommissioning	Delay initially estimated to 30 years	Immediate	Immediate	Immediate	Immediate
Treatment of exotic sodium components (cold traps, cesium traps)	Not treated, will be treated with PX decommissioning	Not treated, stored with metallic sodium	Envisaged to be treated with adapted WVN process	On site treatment by thermolysis, then in situ treatment with WVN process	Best strategy to be determined. Looking at the experimental feedback

8. PROPOSITION OF RECOMMENDATIONS

From this experimental feedback, Table 4 is proposing recommendations for an optimal decommissioning program. Table 4 shows that several options could bring cost reductions. Today a maximum cost reduction of a factor 2 is roughly estimated. Solutions proposed in Table 4 are strongly linked with the ‘classical SFR design’ or what we could consider as GEN I SFR: pool type or loop type with secondary sodium loops. It seems that major gains can only be envisaged if there is a real change in the SFR GEN IV design allowing major simplifications in decommissioning operations.

TABLE 4. STRATEGY FOR AN OPTIMAL DECOMMISSIONING PROGRAM

Macro – tasks	**Optimized decommissioning solution**
Radiological inventory of components and coolant	Must be done in advance. The general trend would be to define metallic material with low activation under neutron flux and small contamination. Special attention must also be paid on coatings and fuel design
Simplification of means and maintenance facilities	Must be carefully prepared in advance. Considerable savings can be done in that field.
Maintaining sodium liquid with other means than primary pumps	This strategy needs to be economically compared with the one consisting in draining all sodium in tanks. Letting sodium in the 1^{ary} vessel can save money if tanks do not already exist. But 1^{ary} sodium treatment remains on the critical path. By draining sodium, new wastes are generated (the tanks themselves) but gains could be done by optimizing and reducing the planning (sodium treatment and primary vessel decommissioning done in parallel). Sodium inside the 1^{ary} vessel is today the preferred option for pool type reactors. If it is the reference option, the facility to keep sodium liquid during defuelling must be defined at early design stage.
Secondary loops draining	Secondary loops and components must be designed to allow good and complete draining. Hot gas sweeping can be a way. Definitive secondary component draining systems (maybe by destructive operations) can be thought at the early design stage (pumps, valves, electro magnetic pumps, etc)
Nuclear material unloading (Fuel and breeder elements)	Design must be done to allow final unloading without dummy core (waste minimization). A good defuelling rate is required. In normal situation an unloading factor reaching 3 assemblies per 24 hours should be achieved. The route for failed assemblies must be defined in advance. Cleaning process should minimize liquid waste effluents and gas consumption. And a unique standardized and efficient process that could treat all assemblies_design would be a great benefit.
Control rods unloading	Design must be done to allow fully drained control rods, event inside the B_4C pellets, or think about other concept. It would be useful, if the same process as for fuel and breeders elements should be applied.
Other assemblies unloading (Steel and reflectors)	Design must integer that all steel assemblies have to be handled with existing handling device defined in normal operation. If not, designers must think how these elements could be remotely withdrawn at end lifetime.
Primary sodium draining	This solution must be economically and technically compared with the sodium storage in the reactor vessel.

Treatment of removable primary big components (PP and IHX)	Cleaning pits must be designed for plant lifetime plus decommissioning estimated time (about 60 years: 40 years operation + 20 years decommissioning). Cleaning pits must be fitted to allow hard decontamination process. Withdrawing components is considered today as the reference strategy. But overall treatment of primary components during the 1^{ary} vessel treatment must be technically and economically evaluated.
Treatment of removable primary small components	Same recommendation as previous.
Treatment of removable secondary component	Due to low activity, this operation could be done in air if a specific area is defined in the early design of SFR. The use of existing cleaning pits could be an issue, the risk is to contaminate Very Low Level Wastes (components from secondary loops) to Low Level Wastes.
Primary sodium treatment	There is today two different industrial process. No major improvement except to think how a same facility could treat several primary sodium coming from different reactors (partially transportable facility).
Treatment of liquid effluents	This part is of major concern because the corresponding wastes are representing high volume, consequently money. An innovative strategy for volume reduction and waste minimization would be very valuable.
Secondary sodium treatment	See ‘Primary sodium treatment’. Another way of development is to analyze how secondary sodium coming from a SFR under decommissioning stage could fill the next SFR under construction.
Complementary primary sodium draining	This operation remains costly. It is mainly due to a lack of decommissioning integration in early design. Major progress can be done to suppress this operation or reduce it to its minimum.
Secondary circuits	Careful check must be done in early design allowing efficient draining of all pipes and components.
Treatment of residual primary sodium remained inside the vessel	Must be done as thorough as possible. Carbonation or WVN are two adapted processes. Therefore both have to industrially demonstrate their ability to safely treat sodium in specific cases: narrow gaps, big heels, etc… Improvements in primary vessel design must be done to: reduce heels of sodium, allow better draining, allow better preliminary sodium treatment, minimize wastes and facilitate future decommissioning operations.
Treatment of residual sodium of secondary loops and their decommissioning	Secondary loops must be designed with a good final drainage. In that case, the preliminary treatment before decommissioning might be reconsidered.
Steam generator treatment and decommissioning	Same remark as ‘Treatment of residual sodium of secondary loops and their decommissioning’.
Primary vessel dismantling	Under water decommissioning is generating large volume of liquid effluents but seems unavoidable to respect dose rate limitation. An issue to this wasted water must be found.
Connection between level 2 and level 3 decommissioning	The sooner seems to be the better. Anyway the planning must be optimized.
Treatment of exotic sodium components	It is one of the major concerns still under development for many SFRs. Cold traps should be rethought to allow their easy decommissioning without minimizing their efficiency in operation.

9. DESIGN OF AN INNOVATIVE SFR CONCEPT – HOW CAN IT EASE THE FUTURE DECOMMISSIONING ?

GENERATION IV program is an opportunity to reconsider from the beginning the concept and design of SFR with some strong innovative options. This situation can be useful to include at the beginning of the project some considerations to reduce future decommissioning costs. Propositions listed below are only personal considerations:

- It is demonstrated for this study that major financing impact on decommissioning SFR remains the sodium treatment in itself as a final waste. Therefore everything that could be to diminish or ease the residual amount of sodium to treat will induce costs saving. Thus, the following improvements can be imagined:
  a. Suppress the intermediate sodium circuits and design a SFR with direct heat transfer between primary sodium and steam production.
  b. Design an innovative SFR with a primary circuit in sodium, in connection with a supercritical CO_2 Brayton cycle.
  c. Design a highly compact primary vessel to minimize total volume of primary sodium.
  d. Simplify component designs to ease their treatment: Steam generators, Intermediate Heat Exchangers, primary pumps…
  e. Imagine some components that could be reused after sodium treatment: cold traps, pumps, modular steam generator, etc…
  f. Consider sodium (secondary but also primary) as a valuable product than can (must?) be reused in future other SFR reactors.
- Another idea to reconsider almost all sodium processes industrial point of view. Up to now, the sodium treatment has found an industrial solution only for the sodium bulk with NOAH process and the DOE process: the process is now well known, industrially operational (3 tons of Na/per day) and efficient. All the other processes – even if they are efficient – remain costly if we compare the time of treatment for the final quantity of sodium treated. Table 5, drawn upon the SPX strategic schedule will highlight this consideration [9].

TABLE 5. PROCESS DEVELOPED FOR SPX AND THE CORRESPONDING AMOUNT OF SODIUM TREATED

Process development	Quantity of sodium treated, ton	Duration years	Ratio: Quantity of sodium treated per year by process type, ton/a
Defuelling of fuel elements	0.2	~ 3	6×10^{-2}
Defuelling of steel assemblies	0.4	~ 2	2×10^{-1}
Dismantling of removable reactors big components (IHX and PP)	1	~ 5	2×10^{-1}
Secondary sodium treatment	2220	~ 1	2220
Primary sodium treatment	3300	~ 2	1650
Reactor vessel carbonation	2.6	~ 1	2.6
Secondary sodium loops carbonation + components	20 (~ 5 tons per loop)	~ 5	4

Values given in Table 5 must be taken with caution because decommissioning time is sometimes not exclusively linked to sodium residual quantity. Neither the less, the values remain significant. Therefore, according to Table 5, efforts must be done to optimize many processes or find new ways to allow the treatment of several sodium families by the use of a common process. The overall treatment of primary vessel with components inside could be one solution, but other ideas have to be proposed. Another option to investigate is to think about some common utilities able to treat sodium wastes of several SFR. This idea was at one time envisaged in France with ATENA concept [10]. In conclusion to reduce significantly the SFR decommissioning costs, solutions and processes will have to be reconsidered thoroughly with new ideas and strategy. This study must be done now in common with early design.

10. CONCLUSIONS

The experimental feedback from actual SFRs is showing that the decommissioning average cost is greater than actual PWRs. Some costs reduction can be envisaged due to actual experimental feedback, process improvements and knowledge transmission. With these considerations, the costs reduction has to be re-estimated, but a maximum gain factor of 2 seems to be a limit. To be competitive with PWRs, innovative SFR concepts must be envisaged to drastically reduce the decommissioning costs. Some technical propositions are made in this paper but a complete investigation has to be considered. The proposition has focused mainly on dismantling operations. They represent 25-30% of the total decommissioning costs. But some other gains must be considered in other items such as waste disposal (25-30% of total cost) and SENEX and Project management. The drastic reduction of decommissioning costs will be feasible only with innovative concept of SFR. Up to now the effort was put to reduce the commissioning costs of future SFRs. Great achievement was proposed in that field. Some innovative studies have reduced SFR construction costs by 4 [11]. The same effort must also integer in the future the decommissioning costs. In parallel it can be recommended to create an expert group providing accurate data from SFR decommissioning costs, to target the lowest decommissioning cost achievable with respect to the following objectives:

- Be easy to proceed and implement;
- Will not interfere with the efficiency of the reactor;
- Will not lessen the inherent safety of the reactor.

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LIST OF PARTICIPANTS

ANDROPENKOV, Sergey	Department of Labour and Environment Protection National Atomic Company ‘Kazatomprom’ 168 Bogenbai Batyr, Almaty, Kazakhstan
ASTEGIANO, Jean-Claude	CEA/Cadarache Reactor Study Department 13108 Saint Paul lez Durance Cedex, France
BAQUÉ, François	CEA/CE Cadarache DER/STR/LTTS Bât. 201 13108 Saint Paul lez Durance Cedex, France
BENSOUSSAN, Erik	PROTEM SA 21 Les Bosses 26800 Etoile-dur-Rhône, France
BERSON, Xavier	CEA/Cadarache DEN/DGI/SEDA Bât 208 13108 Saint Paul lez Durance Cedex, France
BERTE, Michel	Framatome ANP 10 rue Juliette Récamier 69456 Lyon Cedex 06, France
BOULET, Alain	CEA/Cadarache DEN/CAD/SCCAD Bât 101 13108 Saint Paul lez Durance Cedex, France
BOURDELOIE, Christian	CEA/Cadarache DEN/DTN/STR Bât 208 13108 Saint Paul lez Durance Cedex, France
BRAET, Johan A.	Belgian Nuclear Research Center SCK•CEN Boeretang 200 2400 Mol, Belgium
BROCKMANN, Klaus	Forschungszentrum Karlsruhe GmbH S-KNK Postfach 3640 76021 Karlsruhe, Germany
CALMEJANE, Patrick	Westinghouse France Sté WESTINGHOUSE Westinghouse Electrique France SA ZI Saint JOSEPH 4100 Manosque, France
CARRA, Olivier	Framatome ANP 10 rue Juliette Récamier 69456 Lyon Cedex 06, France
DELUGE, Myriam	CEA Marcoule VRH/DDCO/SDSP BP17171 30207 Bagnols sur Céze, France

FARABEE, Oliver Alton Jr
Richland Operations
P.O. Box 550
Richland, WA 99336
United States of America

FONTAINE, Jean
CEA Marcoule
DDCO/SDSP
BP17171
30207 Bagnols sur Céze, France

FOURNIE, Jean-Luc
Framatome ANP
10 rue Juliette Récamier
69456 Lyon Cedex 06, France

GASTALDI, Olivier
CEA/Cadarache
DEN/DTN/STR Bât 208
13108 Saint Paul lez Durance Cedex, France

GIRAUD, Maurice
Framatome ANP
10 rue Juliette Récamier
69456 Lyon Cedex 06, France

GOEDERT, Nicolas
CEA/Cadarache
DTN/STR/LTTS Bât. 201
13108 Saint Paul lez Durance Cedex, France

GOUBOT, Jean-Marc
CEA-VALRHO
DDCO/SDSP
BP17171
Bagnole sur Céze, France

GOUHIER, Eric
CEA/Cadarache
DEN/DGI/SEDA Bât 208
13108 Saint Paul lez Durance Cedex, France

GOUX, Dominique
CEA Valrho
Centrale PHENIX Bât 306
BP17171
30207 Bagnols-sur-Cèze Cedex, France

GRON, Gilles
CEA/Cadarache
DEN/DGI/SEDA Bât 208
13108 Saint Paul lez Durance Cedex, France

GOODMAN, Lynne S.
Detroit Edison Company
Fermi 1
6400 N. Dixie Hwy.
Newport, MI 48166, United States of America

GUNN, James Bain
UKAEA
D2003/Zone 11
Dourneay, Thurso
Caithness KW14 7TZ
United Kingdom

ITO, Chikara
Reactor Technology Section
Experimental Reactor Division, Irradiation Center
O-arai Engineering Center
Japan Nuclear Cycle Development Institute
C/o Centrale Phénix, BP 17171
30207 Bagnols-sur-Cèze Cedex, France

JOULIA, Eric
EDF/CIDEN
35-37 Rue Lois Guerin
BP 1212
69611 Villeurbanne Cedex, France

KIM, Byung Ho
Korea Atomic Energy Research Institute (KAERI)
P.O. Box 105, Yu-sung
Daejon, 305-253, Republic of Korea

KIM, Young-In
International Atomic Energy Agency
Division of Nuclear Power
Wagramer Strasse 5
P.O. Box 100, 1400 Vienna, Austria

LATGE, Christian
CEA/Cadarache
DEN/DTN Bât 710
13108 Saint Paul lez Durance Cedex, France

LERAT, Vincent
CEA Marcoule
VRH/DDCO/SDSP
BP17171
30207 Bagnols-sur-Cèze Cedex, France

LIGER, Karine
CEA/Cadarache
DEN/DTN/STR Bât 208
13108 Saint Paul lez Durance Cedex, France

MAGUIN, Jean-Claude
CEA/Cadarache
DEN/DTN/STR Bât 208
13108 Saint Paul lez Durance Cedex, France

MICHELBACHER, John A. (Bert)
Idaho National Laboratory
P.O. Box 1625
Idaho Falls
ID 83415-6182
United States of America

MINGES, Jürgen Hermann
Forschungszentrum Karlsruhe GmbH (FZK)
S-KNK
Postfach 3640
76021 Karlsruhe, Germany

NOKHAMZON, Jean-Guy
CEA/DEN/DPA CEN Saclay Bat. 121
91191 Gif sure Yvette Cedex, France

PONCET, Bernard
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