IAEA-TECDOC-1405

Operational and decommissioning experience with fast reactors

Proceedings of a technical meeting held in Cadarache, France, 11–15 March 2002

IAEA
International Atomic Energy Agency

August 2004

IAEA-TECDOC-1405

Operational and decommissioning experience with fast reactors

Proceedings of a technical meeting held in Cadarache, France, 11–15 March 2002

International Atomic Energy Agency

August 2004

The originating Section of this publication in the IAEA was:

Nuclear Power Technology Development Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Vienna, Austria

OPERATIONAL AND DECOMMISSIONING EXPERIENCE WITH FAST REACTORS
IAEA, VIENNA, 2004
IAEA-TECDOC-1405
ISBN 92–0–107804–8
ISSN 1011–4289

Printed by the IAEA in Austria
August 2004

FOREWORD

The fast reactor, which can generate electricity and breed additional fissile material for future fuel stocks, is a resource that will be needed when economic uranium supplies for the advanced water cooled reactors or other thermal-spectrum options diminish.

Further, the fast-fission fuel cycle in which material is recycled offers the flexibility needed to contribute decisively towards solving the problem of growing ‘spent’ fuel inventories by greatly reducing the volume of high level waste that must be disposed of in long term repositories. This is a waste management option that also should be retained for future generations.

The fast reactor has been the subject of research and development programmes in a number of countries for more than 50 years. Now, despite early sharing and innovative worldwide research and development, ongoing work is confined to China, France, India, Japan, the Republic of Korea, and the Russian Federation. Information generated worldwide will be needed in the future. Presently, it is in danger of being lost — even in those countries continuing the work.

Many Member States have repeatedly underscored the importance of knowledge preservation and transmission to the young generation. 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. Within the framework and drawing on the wide expertise of its Technical Working Group on Fast Reactors (TWG-FR), the IAEA convened, November 22–24 1999, an Advisory Group Meeting on Evaluation of Fast Reactor Core Physics Tests. The outcome of this advisory group meeting was to underline the importance of an IAEA initiated effort to preserve fast reactor physics knowledge with an exhaustive review of all available sources of fast reactor experimental data, including data from critical facilities.

The initiative received further support at a special technical session, entitled Passing on Fifty Years of Fast Reactor Knowledge to a New Generation in Nuclear Research and Development, which was held at the 2001 American Nuclear Society Winter Meeting, and co-chaired by Argonne National Laboratory (ANL) and IAEA staff.

In another, more general development, the IAEA Director General convened a meeting of senior officials form around the world to address the more general issue of nuclear knowledge management (17–19 June 2002). At this meeting, there was widespread agreement that, for sustainability reasons, long term development of nuclear power, as a part of the world’s future energy mix, will require fast reactor technology and, that given the decline in fast reactor development projects, data retrieval and knowledge preservation efforts in this area are of particular importance.

Operational experience constitutes an important aspect of any fast reactor knowledge base. It is within the framework of the aforementioned initiative, and with the objective of gathering and exchanging information on fast reactor operational and decommissioning experience, and initiating the process of analyzing it (a necessary step in the process of transforming ‘knowledge’ into ‘wisdom’), that the IAEA convened the Technical Meeting on Operational and Decommissioning Experience with Fast Reactors. The technical meeting, hosted by CEA, Centre d’Etudes de Cadarache, France, was held 11–15 March 2002.

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 A. Stanculescu of the Division of Nuclear Power.

EDITORIAL NOTE

CONTENTS

SESSION 2: FAST REACTOR DECOMMISSIONING EXPERIENCE

SESSION 3: FAST REACTOR PHYSICS AND ENGINEERING EXPERIMENTS AND ANALYSES

SESSION 4: PRESERVATION OF FAST REACTOR KNOWLEDGE AND EXPERIENCE

SUMMARY

1. INTRODUCTION

For three decades, several countries had large and vigorous fast breeder reactor development programmes. In most cases, fast reactor development programmes were at their peaks by 1980. Fast test reactors [Rapsodie (France), KNK-II (Germany), FBTR (India), JOYO (Japan), DFR (United Kingdom), BR-10, BOR-60 (Russian Federation), EBR-II, Fermi, FFTF (United States of America)] were operating in several countries, with commercial size prototype reactors [Phénix, Superphénix (France), SNR-300 (Germany), MONJU (Japan), PFR (United Kingdom), BN-350 (Kazakhstan), BN-600 (Russian Federation)] just under construction or coming on line. From that time onward, 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 shutdown — with EBR-II permanently, and FFTF in a standby condition. Thus, 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 scheduled to be dismantled by 2004; in the UK, PFR was shut down in 1994; BN-350 in Kazakhstan was shut down in 1998.

It is difficult to argue that fast breeder reactors will be built in the near term when no commercial market exists and there is a plentiful supply of cheap uranium. Nevertheless, it is reasonable to assume that, were nuclear energy to remain an option as part of the long term world energy supply mix, meeting the sustainability requirements *vis-à-vis* natural resources and long lived radioactive waste management will require deploying systems involving several reactor types and fuel cycles operating in symbiosis. Apart from cost effectiveness, simplification, and safety considerations, a basic requirement to these reactor types and fuel cycles will be flexibility to accommodate changing objectives and boundary conditions. This flexibility can only be assured with the deployment of the fast neutron spectrum reactor technology, and reprocessing.

At the same time that the interest in the fast reactor waned, the retirement of many of the developers of this technology reached its peak, between 1990 and 2000, and hiring diminished in parallel. Moreover, R&D programmes are being discontinued, and facilities falling in disuse. Under these circumstances, the loss of the fast reactor knowledge base should be taken seriously. One particularly important aspect of this knowledge base is given by the accumulated operational experience.

The participants in the 33rd Annual Meeting of the International Working Group on Fast Reactors, Technical Committee Meeting on Liquid Metal Fast Reactor Developments (Vienna, 16–18 May 2000), recommended holding a technical meeting (TM) on Feedback from Operational and Decommissioning Experience with Fast Reactors.

At the 34th Annual Meeting of the Technical Working Group on Fast Reactors, Technical Committee Meeting on Review of National Programmes on Fast Reactors and ADS (Almaty/Kurchatov City, Kazakhstan, 14–18 May 2001), it was further recommended to launch a Co-ordinated Research Project (CRP) on Generalization and Analyses of Operational Experience with Fast Reactor Equipment and Systems (Preserve Fast Reactor Operation and Decommissioning Experience). It was agreed to structure the TM in such a way that, apart from providing an information exchange opportunity, it would also prepare the grounds for the CRP.

2. SCOPE AND OBJECTIVES

The scope of the TM was to provide a global forum for information exchange on fast reactor operational and decommissioning experience.

The objectives of the TM were to:

- Exchange detailed technical information on fast reactor operation and/or decommissioning experience with DFR, PFR (UK); KNK-II (Germany); Rapsodie, Phénix, Superphénix (France); BR-10, BOR-60, BN-600 (Russian Federation); BN-350 (Kazakhstan); SEFOR, EBR-II, Fermi, FFTF (USA.); FTBR (India); JOYO, MONJU (Japan);
- Present the status of the work concerning the knowledge preservation efforts related to the experience accumulated in the various member states from the operation and decommissioning of fast reactors;
- Start the preparation of the planned Co-ordinated Research Project (CRP) on Generalization and Analyses of Operational Experience with Fast Reactor Equipment and Systems (narrow down scope and objectives of the CRP, propose a detailed work plan).

3. GENERAL STATEMENTS

3.1. France

3.1.1. Background

In France, the first chapter in the history of fast reactors was the construction of the mixed oxide fuelled, sodium cooled ‘Rapsodie’ reactor (1962–1966). The operation of Rapsodie was excellent from 1967 to 1978 (initially at 24 MW(th), it was upgraded to 40 MW(th) in 1971). Rapsodie was an outstanding irradiation tool, allowing the demonstration of oxide fuel capabilities, and an initial screening of the core structural material. However, from 1978 to 1982, the detection of primary sodium aerosols in areas surrounding the primary circuit disturbed its operation. The reactor was finally shut down in April 1983, after several end-of-life tests; at this time, Phénix had proved able to ensure all irradiation needs. Since 1983, the reactor has been undergoing decommissioning. The objective is to reach the IAEA level 2 by 2005, a surveillance state should then last from 2005 to 2020 before final decommissioning.

The prototype fast reactor, Phénix (a pool type reactor, 250 MW(e)) went into commercial operation in 1974. To date, 51 cycles were run and more than 20 billion KW•h were produced. As the initial lifetime of the reactor was 20 years, the reactor should have been shut down in 1994, but in the mid-nineties, the role of the reactor changed: it was to be used as an irradiation tool acting as a support to CEA’s R&D transmutation programme within the framework of the 1991 French law concerning long lived radioactive waste management. This new objective required an extension of the planned reactor lifetime. A large renovation programme was defined, and today most of this renovation programme has been accomplished. The greater part of the work still underway concerns repairs on the steam generators. Resuming power is planned for the 1st half of 2003, with a total of 6 operating cycles to carry out the experimental irradiation programme. The overall period will cover about 5½ years.

Construction of the SuperPhénix plant lasted from 1977 to 1985. Full power was reached in 1986, and until the end of 1996, the plant operated for 4½ years at different levels of power, with scheduled periods of maintenance and tests. It remained shutdown for 4½ years, although still in an operational state, due to ongoing administrative procedures, and a little more than 2 years shutdown were due to technical incidents and repairs. The last operating year was remarkable: the complete programme of overall qualification by successive stages to 30, 60, and 90% nominal power progressed without difficulty. After an interruption of activity of more than 5 years, all the parameters were found to be normal. However, following the declaration made to the French National Assembly on June 19, 1997, the French government decided on February 2nd 1998, to permanently shut down the SuperPhénix plant.

The governmental decree of 31st December 1998 finalized the immediate and permanent shutdown of the plant. In conformity with this decree, the following operations were carried out:

- Core unloading from the reactor vessel and transfer to the fuel storage pool. By the end of February 2002, all 358 fuel elements had been transferred, as well as almost half the breeders and part of the control rods. Out of a total of 650 elements, 480 have already been unloaded;
- Removal from service of non-required systems;
- Studies for primary vessel draining and sodium treatment.

In 2001, EDF made the decision to dismantle all first generation reactors, including SuperPhénix up to the IAEA level 3 by the year 2025, without intermediate safe storage status (level 2). Phénix will remain operational, as mentioned before, through 2008–2009.

3.1.2. Ongoing sodium cooled fast reactor R&D

3.1.2.1. Preservation of acquired knowledge

Considerable effort has been put into long term knowledge preservation (storage and access) for use in future sodium cooled fast reactor designs. This task involves various activities: the elaboration of synthesis reports (including the SIMMER validation and CABRI experiment synthesis for CDA analysis), SuperPhénix data storage (operational feedback), Phénix lifetime extension feedback (in-service inspection and repair), updating of neutronics data banks and code validation efforts, the new edition of RCC-MR (material analysis rules and criteria) and RAMSES2 (irradiated structural material rules).

Preserving acquired knowledge not only includes feedback obtained from the Rapsodie, Phénix, and SuperPhénix reactors, but also knowledge acquired at the time of the EFR Project (1988–1998) that allowed considerable improvements to be made after careful observation of the SuperPhénix reactor in terms of technology, in-service inspection, safety, steam generator design, and neutronics.

3.1.2.2. Irradiation programme

Experiments in the Phénix reactor (materials, transmutation of actinides, and irradiation of targets containing long-lived fission products), and in BOR-60 (transmutation of americium, nitride fuels).

3.1.2.3. Dismantling

A considerable amount of R&D has been launched to solve not only the problems encountered during the dismantling of SuperPhénix (e.g. assembly washing, draining of the reactor block, treatment of residual sodium), but also those encountered in the dismantling of experimental equipment used over the past years (the treatment of experimental devices taken from the irradiation reactor, SILOE using Na-K, technological test devices, and so forth). And finally, preliminary studies have been carried out in a new radioactive sodium waste treatment facility enabling us to process waste stored in the various CEA centres.

3.1.2.4. Maintaining the CEA expertise

CEA wishes to maintain its expertise in the field of liquid metal reactors, a competence which today is based on a R&D programme covering four decades of studies. At present, following the shutdown of the SuperPhénix reactor, our efforts in terms of R&D have been greatly reduced, but maintaining our expertise will allow us, well beyond the support provided to the operation of the Phénix reactor, to exchange knowledge with other countries pursuing R&D in the field of liquid metal fast reactor technology. This exchange of experience, acquired over a period of 25 years of operation of the Phénix reactor, operation of the SuperPhénix (on a commercial scale), and the EFR studies, along with the studies described above, will allow us to sustain extensive knowledge about the liquid metal reactor type, and thus enable us to evaluate further developments.

In conclusion, although the CEA has now decided to focus its R&D on gas cooled reactor concepts, with the prospect of perfecting a gas cooled fast reactor in the long term (4th generation), it nevertheless shall preserve activities and expertise it has acquired on sodium cooled fast reactors over the years. CEA would like to pursue further exchanges with other countries who are also engaged in liquid metal fast reactor research.

3.2. Germany

In Germany, activities related to the development and operation of fast breeder reactors have been terminated.

The SNR 300 power reactor was not taken into operation. All sodium-wetted components were removed. The inactive sodium was disposed off. The fuel elements that had already been fabricated for the first core were put into containers, welded gas-tight, and stored at the federal nuclear fuel storage facility.

The compact sodium cooled reactor facility KNK (20 MW(e)) was shut down in 1991 and currently is in the state of advanced dismantling. The nuclear fuels and sodium were disposed off. All systems were removed. Disassembly of the activated reactor tank and its internals has been started. Complete dismantling of the reactor building is planned to be completed by 2005.

3.3. India

The present status with respect to the fast breeder test reactor operating experience and prototype fast breeder reactor design are presented below.

3.3.1. Fast breeder test reactor

The fast breeder test reactor (FBTR) is a loop type reactor located at Kalpakkam, India. The reactor has been operated for 27 600 h till now at various power levels up to 17.4 MW(t). The peak burn up of 90 000 MW•d/T was achieved in the 70% PuC + 30% UC Mark-I fuel. Turbo Generator was synchronized to the grid with the nuclear steam to check its performance. Continued operation of TG is planned at high power. The post irradiation examination of the fuel discharged after 50 000 MW•d/T peak burnup showed adequate gap between fuel and clad indicating higher burn up is possible. Present linear heat rating is 400 W/cm and the target burn up of the fuel is to be increased to 100 000 MWd/T. The core has 35 sub-assemblies (SA) now.

A reversible reactivity transient, which sets in at a specific core ΔT was observed during low power operation. The magnitude of the transient reduces and the ΔT at which the reactivity transient sets in increases with respect to increase in core flow. The reactor operation was continued above a primary flow of 450 m^3/h, as the phenomenon does not occur above this value.

The core cover plate mechanism, which supports core thermocouples, got stuck at a higher position and could not be lowered. The safety implication of the above was analyzed and thresholds of core ΔT and core mean temperatures were lowered and reactor operation continued.

The effectiveness of delayed neutron detectors for detecting clad failure was tested by operating the reactor with vented fuel SA in the core. The void coefficient of reactivity at various core locations were measured using two special SA fabricated for this purpose. The void coefficient was found to be negative.

Comprehensive radiation survey has been carried out to ensure shielding efficiency in the cells having primary radiation components and cover gas system. The measured dose rates have been found to be less than the design values. The average annual collective dose is 2.2 P-mSv. This indicates very low radiation exposure from the plant.

Performance of FBTR till now has been very good and all problems encountered initially have been overcome. The experience gained in operating the reactor is valuable. It is planned to expand the core to full size in near future.

3.3.2. Prototype fast breeder reactor

The Prototype fast breeder reactor (PFBR) is a 500 MW(e), pool type sodium cooled reactor with 2 primary pumps, 4 intermediate heat exchangers and 2 secondary loops with 4 steam generators per loop. The detailed design, R&D, manufacturing technology development and safety review are nearing completion. Major engineering experiments with respect to thermal hydraulics, component testing, sodium technology etc. have been completed. To demonstrate technology development, full scale model/scaled down model/sector model of components such as primary sodium pump, steam generator, reactor vessel, roof slab, control rod drive mechanisms etc. have been fabricated. In order to design an optimum in-vessel shielding, a series of fast reactor shielding mock up experiments involving transport through typical shield configurations of steel, sodium, graphite and boron carbide have been carried out. Radiation streaming mock up experiments are also planned. PFBR will be constructed at Kalpakkam and the detailed project report for this project will be submitted for sanction shortly.

3.3.3. Knowledge preservation

India is in the initial stages of the commercialization of the fast reactors and the efforts on knowledge preservation goes in parallel with the design and operation. Care is taken to ensure that all design and operation data are documented and archived with proper identification. In FBTR all design, drawings and operation related documents are stored separately in an air-conditioned record room for immediate and future reference. However, for PFBR, from the beginning all design documents and drawings are maintained in the electronic form. It is also planned to provide access to all these documents in the Intranet at IGCAR through password protection. Utilizing the experience gained by the experienced manpower over the years is an important aspect in knowledge preservation. This is planned to a limited extent by involving the knowledgeable people by participating in seminars, training courses etc. Efforts are also on to introduce courses related to nuclear energy in colleges and sponsoring new engineers/scientists to these courses. To consolidate the knowledge available internationally, an effective way would be to post the documents regarding the operating experience of all fast reactors in electronic form and make them available to all fast reactor specialists.

3.4. Japan

JNC is undertaking a major program of research and development on liquid metal cooled fast breeder reactors, which is fully supported by the government of Japan and the electrical utilities. Hence, the perspective of JNC on knowledge preservation is rather different from that of organizations where the fast reactor project has been scaled down or discontinued. Within JNC, there is a statutory obligation to preserve documentary records of the fast reactor project. Over time the method of archiving has changed from optical (microfilm, microfiche etc.) to digital storage. It is the long term objective of JNC to convert all its records to digital format and make them available to staff over its intranet. JNC is also attempting to preserve ‘human knowledge’, that is, the expertise of staff who have been involved in the fast reactor project over a long period and who are now nearing retirement. Based on this information, two computerized systems are currently being constructed: one which records in a readily accessible manner the background to key design decisions for the Monju plant; and a second which uses simple relationships between design parameters to aid designers understand the knock-on effects of design choices (joint project with Mitsubishi).

To its partners in international cooperation — the US/DoE and the organizations of the Euro-Japan collaboration — JNC is proposing a joint approach to knowledge preservation and retrieval. The proposed concept, dubbed the International Super-Archive Network (ISAN), would make use of the standardized software the new technologies of the Internet increase the mutual accessibility of fast reactor information.

JNC considers it extremely important to reflect the lessons learnt from previous experience in the fast reactor field to the operation and maintenance of Monju and the design of future reactors.

3.5. United States of America

All first generation fast breeder reactors have been shutdown or decommissioned with only the fast flux Test facility remaining in a standby condition awaiting final decommissioning. Fast breeder reactor development activities have been terminated with limited technology development in transmutation and pyroprocessing.

The EBR-II, 62MW(t), has completed decommissioning steps and is now in a radiological and industrially safe condition at the direction of the US DOE. These activities followed removal of fuel for conditioning and the disposal of all bulk sodium coolant. As required by US regulations, residual sodium within reactor system will be deactivated under appropriate environmental permits after which the EBR-II will await future dismantling.

4. SESSION SUMMARIES

4.1. Session 1: Sodium cooled fast reactor operational experience

The papers presented a comprehensive overview of the accumulated experience with the operation of sodium cooled fast reactors. The worldwide 40+ years of fast reactor development represent a total of 300 years of operation. Based on this figure, it was concluded that the sodium cooled fast reactor technology has reached a mature stage. The advantages of this type of reactor were pointed out by the various presenters:

- Safe and reliable operation;
- Easy operation and maintenance;
- Low environmental impact;
- Demonstration of fuel cycle closure in some cases;
- Flexibility for fuel cycle issues.

The technical difficulties encountered during the operation of fast reactors, and their resolution, were presented. The status of fast reactor development in the different countries is currently in a wide range:

- Reactors being decommissioned;
- Operating reactors in a lifetime extension process;
- Reactors under construction or in the commissioning phase.

Several prototype reactor projects are going ahead, e.g. in China, India, Japan, the Russian Federation (these reactors are likely to be commissioned by 2010). However, large scale commercial reactor construction is not expected before 2020. There is a major interest for all countries to preserve the operational experience for both the ongoing and future long term projects.

4.2. Session 2: Sodium cooled fast reactor decommissioning experience

Decommissioning experience (both direct experience and decommissioning planning activities) with sodium cooled fast reactors was presented in contributions from France, Germany, India, Kazakhstan, and the USA. The discussions centred on common technologies and consistency of approach. Noted differences are justified on the grounds of regulatory requirements rather than differences in technologies. The resulting conclusions and recommendations were:

1) Advanced planning is essential;
2) Remove fuel as soon as possible;
3) Proceed with proven technologies;
4) Proceed quickly;
5) Move from secondary to primary (from less active to more active);
6) Primary systems dismantling require specialized techniques and expertise (e.g. remote technologies);
7) Maintain operational staff, utilize contractors familiar with the plant, minimize total staff;
8) Maintain interaction with regulators at all times;
9) Potential topics for R&D are:
    - Sharing of important information;
    - Advanced planning tools;
    - Sharing of technologies, e.g.

i. Specific techniques (carbonation, further sodium draining);
ii. Specific applications (de-fuelling, dismantling of secondary systems);
iii. Available advanced sodium removal technologies;

- Feedback for future reactor designs, e.g.
  i. Application of ^{60}Co dose rate minimization measures;
  ii. Design of sodium draining systems and components with a specific view to decommissioning.

4.3. Session 3: Fast reactor physics and engineering experiments and analyses

The technical session on fast reactor physics and engineering experiments and analyses focused on some R&D performed in experimental and power fast reactors. The work done at CEA to understand the failure due to heat affected zone stress relief or reheat cracking in austenitic stainless steel welds, particularly in stabilized 321 or 347 materials working at high temperatures was reported. In the discussions, it was brought out that 321 steel is more suitable for low temperature applications, wherein such difficulties were not experienced. RCC-MR code does not recommend usage of this material for high temperature applications. It was suggested that this study will be useful for life time extension of reactors wherein such steels are in use. The status of the RCC-MR code was presented by Framatome, ANP. The new edition of the RCC-MR code will be brought out in French and English language shortly. It was brought in the discussions that this code will find wide usage in high temperature nuclear reactor design and also in other high temperature systems.

CEA's presentation covered neutron physics commissioning experiments for Superphénix and Phénix, which were re-evaluated using the recent ERANOS-1.2 code system. In the discussion, the following points were mentioned: The misprediction of decay component of burn-up reactivity swing needs to be investigated in the view points of use of higher neutron energy groups, or improved fission products nuclear data. Accurate modelling process should be recommended especially for control rod worth calculation, whereas 2D homogeneous modelling gives close results with 3D for other parameters as criticality mass. The ERANOS system is applicable to other type of fast reactor (e.g. gas and heavy liquid metal cooled), but the data is not validated for this applications.

The results of physics and engineering experiments in the fast breeder test reactor (FBTR) were presented as well. It was brought out that it is essential and mandatory to carryout important safety related physics and engineering tests to validate the data used in safety evaluation. The feedback in these experiments were also used to validate and redefine various mathematical models/codes for better prediction.

4.4. Session 4: Sodium cooled fast reactor knowledge preservation

France, Japan and the Russian Federation presented the status on sodium cooled fast reactor experience preservation made in these countries. The reports underlined these countries' large experience with design, construction and operation of sodium cooled fast reactors. The discussions underlined the importance of the IAEA support for knowledge preservation of fast reactor experience.

Recognizing the importance of sodium cooled fast reactor knowledge preservation, there is the need to pursue an internationally coordinated activity aiming at the analysis of operational experience with fast reactor equipment and systems, as well as at the generalization of the lessons learned. The objectives of this activity are to:

- Safeguard the feedback from commissioning, operation, and decommissioning experience of experimental and power sodium cooled fast reactors;
- Enable easy access to the information from this feedback;
- Attempt at generalization/synthesis of lessons learned from the commissioning, operation, and decommissioning of experimental and power sodium cooled fast reactors,

and its main tasks are to:

- Establish the list of the reactors to be considered;
- Define/agree on topical areas;
- Establish the catalogue of documents and references to be included;
- Define the structure of the abstracts, and the format of the references;
- Produce key words glossary for the various topical areas;
- Define the path for sequential searches for the various topical areas;
- Establish the structure of the database and define the rules for access, sharing etc. (e.g. define several levels of access);
- Produce a synthesis/generalization of commissioning, operational, and decommissioning experience.

As a next step, the need to establish an international sodium cooled fast reactor database was put forward. In an ad-hoc working group, the session elaborated the general structure of such a database. Table 1 gives an example of the structure.

TABLE 1. STRUCTURE OF A DATABASE

Fields	Example	Remarks
Part – A		
Title	R&D LMFRs Knowledge Preservation French Project	
Authors 1	F. Baqué	
Affiliation 1	CEA, Cadarache, France	
Authors 2		
Affiliation 2		
Report number		
Key words 1 – Sequential		To be defined by technical administrator
Key words 2 – Random	Liquid metal fast reactor, LMFR, knowledge preservation, CEA, safety, working thermohydraulics, nuclear fuel	
Abstracts		
Paper text		
Paper format		Pdf
Date of publication	11.03.2002	
Part – B		
USI number		To be defined by technical administrator
Server number		Server identification where full paper is stored
Contributing organisation	CEA, Cadarache, France	

Part A of the structure deals with fields specific to the publication and the technical content of the paper, whereas Part B deals with fields required for the maintenance of the report by technical and system administrator.

The database must be designed such that it is available to the users around the world through Internet. To enable this, a suitable search facility shall be included. Searching shall be possible on specific fields such as title, author, affiliation, report number, keywords and abstracts. Restricted searching such as published during the period (from, for example 1.1.2001 to 31.12.2001) shall also be possible.

For large technical databases, sequential searching on specific, pre-defined key words will be useful. These keywords sequence shall be pre-defined by the technical administrator group and shall be assigned to each data. This keyword sequence may also be converted to a number similar to US1 classification of books. Each report can have more than one USI numbers depending on the contents. For searching reports in this way, the user has to continue search by selecting the fields from the pull down menu. An illustration is given in Fig. 2.

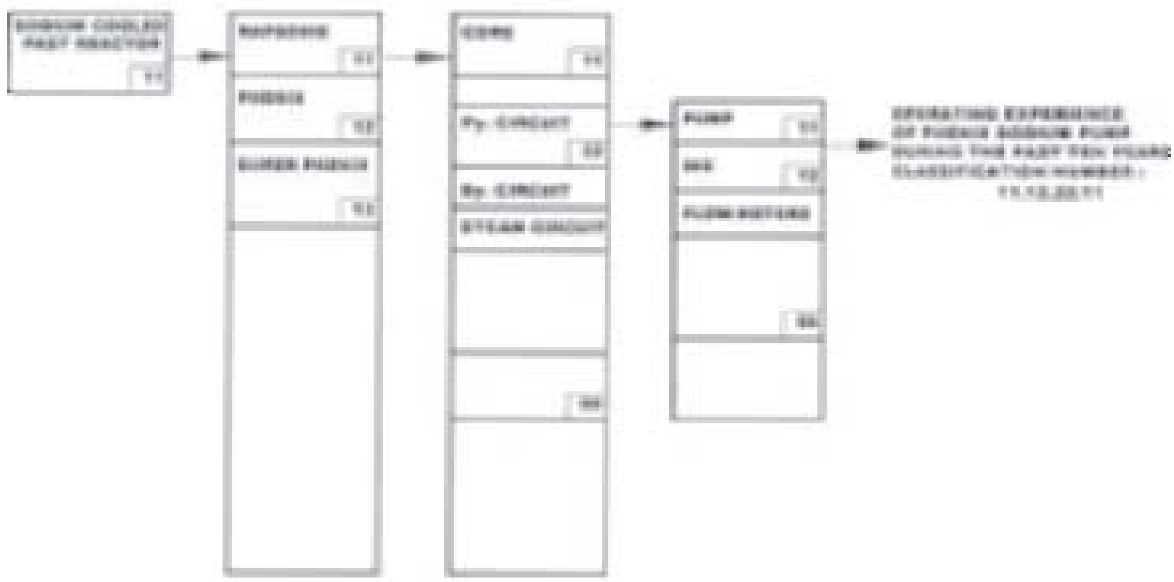


FIG. 2. Sequential searching.

The technical contents of the database shall include information from various published journals, papers presented in seminars, etc. Also classified information from specific organizations may also be placed on the database with restricted access. Hence the design shall consider provisions such as: open literature — accessible by all, information available to specific organizations, on payment through on line credit card, payment through off line procedures, to specific organizations by following specific procedures, etc.

The database shall be defined such that the data can be uploaded from specific organizations, through password control. This will enable uploading of data from various member countries. A suitable field in the database shall be available to keep track from where the data was uploaded. Additionally provision shall be available for any individual to post complete data (literature/report, etc.) through the Internet. This will be scrutinized by the technical administrator and uploaded to the database. However the allotment of a USI number shall remain with technical administrator.

A server located in a centralized place shall contain major information. It shall contain all the fields in the structure, but with or without full text of reports. This server shall contain the information that are required for execution of a search and the results are passed on to the user through Internet. Subsequently when the user asks for full text of the paper, it shall be made available to them. For storage of full text of reports, the servers containing the full text may remain distributed in the member countries. This distributed system will enable users to get all the information and at the same time satisfy the countries having classified information, the information is kept restricted and is available with them only. A pictorial representation of this arrangement is shown in Fig. 3.

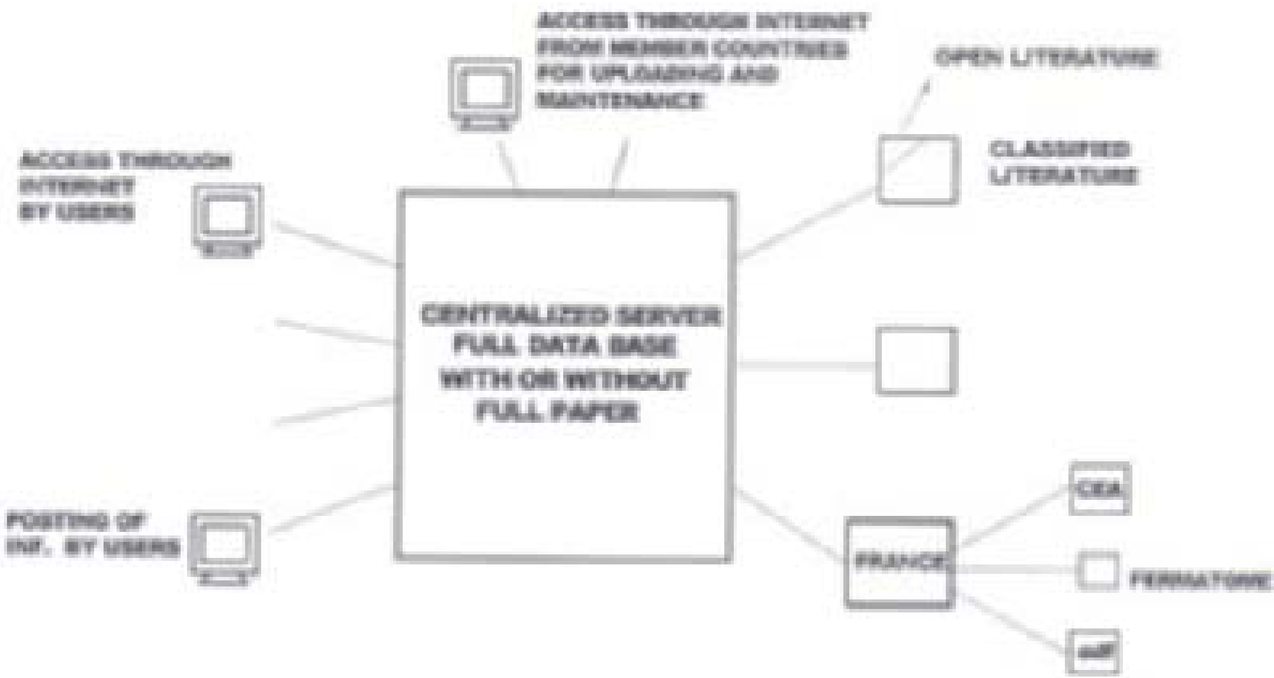


FIG. 3. Hard ware organization.

The establishment of such a database is an ambitious and time-consuming process. The implementation of such a task can be envisaged in the three phases:

1) Development and testing:
   This is a process that can be planned and completed in a reasonable time of $\simeq$ 18 months. During this period the complete architecture of the software and hardware shall be completed. The sequential searching keyword structure shall be established. The server architecture, including establishment of links to member countries can be considered. Also minimum amount of data shall be posted in the servers and tested for its functioning.

2) Uploading data and making database available to public:
   Existing information such as those in INIS Atomindex, journals etc. shall be uploaded. The organization responsible for maintenance of the full text distributed in different countries shall be defined and servers shall be made available.

3) Continuous maintenance and upkeep of database:
   Maintenance is essential so that the database is made available 24 hours per day. Also care shall be taken from development stage onwards such that the developments in the electronic media will always assist in the maintenance of the database.

FAST REACTOR OPERATIONAL EXPERIENCE

(Session 1)

FAST BREEDER TEST REACTOR. 15 YEARS OF OPERATING EXPERIENCE

K.V. SURESH KUMAR, R.P. KAPOOR, P.V. RAMALINGAM, B. RAJENDRAN, G. SRINIVASAN, K.V. KASIVISWANATHAN
Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, India

Abstract

Fast Breeder Test Reactor (FBTR) is a 40 MWt/13.2 MWe sodium cooled, loop type, mixed carbide-fuelled reactor. Its main aim is to gain experience in the design, construction, and operation of fast reactors including sodium systems and to serve as an irradiation facility for development of fuel and structural materials for future fast reactors. It achieved first criticality in October 1985 with Mark I core (70% PuC - 30% UC). Steam generator was put in service in January 1993 and power was raised to 10.5 MWt in December 1993. Turbine generator was synchronized to the grid in July 1997. The indigenously developed mixed carbide fuel has achieved a peak burnup of 88 000 MWd/t till now at a linear heat rating of 320 W/cm and reactor power of 13.4 MWt without any fuel-clad failure. The paper presents operating and decontamination experience, performance of fuel, steam generator, and sodium circuits, certain unusual occurrences encountered by the plant and various improvements carried out in reactor systems to enhance plant availability.

1. INTRODUCTION

Fast Breeder Test Reactor (FBTR) at Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam is 40 MWt/13.2 MWe sodium cooled, loop type plutonium rich mixed carbide fuelled reactor. It has two primary and secondary loops and a common steam water circuit with once through steam generator (SG), which supplies super heated steam to the condensing turbine.

There are two SGs per loop and are located in a common casing. The SGs are not insulated to facilitate decay heat removal by natural convection of air the casing. A 100% steam dump facility is provided in the steam water circuit so as to operate the reactor at full power for experimental purposes even when turbine is not available. The basic conceptual design of block pile, primary loop and reactor instrumentation is similar to French reactor Rapsodie, whereas steam-water circuit and turbo generator (TG) were designed indigenously. The major design parameters of the reactor are given in Table 1.

TABLE 1. MAIN CHARACTERISTICS OF FBTR

Reactor power		40 MWt/13.2 MWe
Reactor coolant		Sodium
Concept of primary circuit		Loop (2 nos.)
Fuel	Mark I	70% PuC + 30% UC
	Mark II	55% PuC + 45% UC
Fuel pin diameter		5.1 mm
No. of pins in a subassembly		61
Control rod material		B_4C (90% enriched in B^{10})
Neutron flux		3 E 15 n/cm^2/s
Core height		320 mm
Reactor inlet sodium temperature		380°C
Reactor outlet sodium temperature		515°C

ctd

Core flow	1100 m^3/h
Secondary sodium flow	300 m^3/h per loop
Steam temperature	480°C
Steam pressure	125 kg/cm^2
Feed water flow	70 T/h
Sodium inventory	150 T
Steam generators	Once through type, 7 tubes in a shell, in triple "S" shape
Turbine generator	16 stages, condensing type 16.4 MWe rating with air cooled generator

The reactor attained its first criticality in October 1985 with Mark I core consisting of 22 fuel subassemblies of indigenously developed fuel (70% PuC + 30% UC). While carrying out low power physics experiment (<500 kWt), a fuel-handling incident took place in May 1987 and the reactor could resume operation only in May 1989 after recovering from the incident. Subsequently, low power physics and engineering experiments up to 1 MWt were completed in 1992. After completion of commissioning of SG and its leak detection system, reactor power was raised to 10.2 MWt in December 1993. After completing high power engineering and physics tests, reactor power at high power was continued.

Construction and commissioning of Turbo-Generator (TG) and its auxiliaries were subsequently completed and TG was synchronized to the grid producing 1.2 MWe in July 1997. Reactor core was gradually enhanced and power was raised in steps to 13.4 MWt. Fuel pins of Mark I and Mark II compositions were irradiated in the reactor and discharged for post irradiation examination (PIE) to assess the fuel performance. Reactor was operated at 8 MWt for irradiation of zirconium-niobium (Zr-Nb) allow for PHWR programme. Nine irradiation campaigns have been completed so far up to a maximum power level of 13.4 MWt and peak burnup of 88 000 MWd/t was achieved without any fuel-clad failure so far. The reactor parameters achieved so are given in Table 2.

TABLE 2. ACHIEVED PARAMETERS (FEBRUARY 2001)

Power	13.4 MWt
Linear heat rating	320 W/cm
Peak burnup	88 000 MWd/t
Total operating time	27.593 h
Total thermal energy developed	138 756 MWt
Primary sodium flow	520 m^3/h
Reactor inlet/outlet temperature	350/430°C
Feed water flow	20 T/h
Steam conditions	420°C at 120 kg/cm^2

It is planned to operate the reactor at 17.4 MWt with peak linear heat rating of 400 W/cm and achieve a burnup of 100 000 MWd/t from March 2002 onwards.

2. OPERATING EXPERIENCE [1, 2]

2.1. Fuel

The indigenously designed and developed mixed carbide fuel (70% PuC + 30% UC) set a record when it was used as driver fuel for the first time in the world [3]. The reactor was initially loaded with a small Mark I core rated for 10.5 MWt at a linear heat rating (LHR) of 320 W/cm. Since the fuel is new, it was felt desirable to ascertain its performance before increasing the reactor power in a phased manner.

Fuel pins of Mark I and Mark II fuel compositions were irradiated and discharged for PIE to assess its performance. Similarly the central fuel sub assembly (FSA) was discharged for PIE after it has reached a target burn up of 25 000 MWd/t. Detailed PIE indicated that although the fuel clad gap was seen to be closing due to cracking of the fuel but still adequate gap was existing in addition to gap available in the fuel due to cracks to accommodate further swelling [4]. Further one of the FSA in the first ring of the core, which has seen a burn up of 50 000 MWd/t, was discharged from the core and PIE was carried out.

Visual examination of the Subassembly and the fuel pins indicated their good health. Dimensional measurement on the SA and fuel pins revealed that there is no significant deformation or distortion either on the hex-can or on the fuel pins. Eddy current testing (ECT) and X-radiography did not reveal any defect in the fuel pin clad. Increase in stack length varied from 4.07 to 5.32 mm (i.e. an average increase in stack length of 1.44%).

Neutron radiography of the pins was carried out for the first time and the results were comparable with X-radiography results. There was no evidence of any abnormality or redistribution of actinides in neutron radiography. The fission gases were extracted and analyzed. The total gas release into the plenum (Xe+Kr) was found to be varying from 6 to 22% for 50 000 MWd/T fuel compared to 1 to 3% for 25 000 MWd/T fuel. Metallography of cut -cross section revealed that no restructuring of fuel had taken place. However reduction in porosity was noticed for 50 000 MWd/T fuel compared to 25 000 MWd/T fuel.

This may be due to internal accommodation of swelling in the fuel matrix. Photo mosaics of the cut cross section of fuel indicated diametric increase due to swelling of the order of 1.77%.

Cracks and sinter porosities still available in the fuel indicate that fuel can undergo further swelling before fuel-clad mechanical interaction can exert stresses on the clad. Microstructure analysis of clad tube did not indicate any carburisation. There was also no significant reduction in hardness clad.

Based on the excellent performance of the fuel in terms of swelling and fission gas pressure build up, the ratings were increased to 400 W/cm LHR and 100 000 MWd/t burnup. At each stage of LHR & burnup enhancement, rigorous theoretical analysis was carried out and safety clearance obtained.

Further, it is planned to expand the reactor core gradually to 76 FSA of Mark II core compositions in the next two years to achieve nominal power operation.

2.2. Sodium systems

Sodium systems have been operating for the past sixteen years and their performance has been excellent. The impurity levels in sodium was always < 0.6 ppm and it was demonstrated that even without purification system in service, the impurity levels in primary system remained within limits. During commissioning of steam generator, one cold trap in secondary sodium loop had to be replaced due to impurity loading at the time of connecting the SG to the loop. One secondary sodium pump was replaced after 10 000 h of operation due to abnormal noise [5]. Performance of all other pumps till now was very good. Performance of sodium pump drive system was not satisfactory initially. It improved significantly after air conditioning the control logic panels and carrying out certain logic modifications. The primary sodium was sampled for trace element analysis and the nuclear grade purity is well maintained.

An electro-chemical carbon meter is installed in one of the secondary sodium loops to measure the active carbon level in the system. Its performance is being studied.

2.3. Reactor assembly

The performance of control rod drive mechanism (CRDM) has been satisfactory with friction force within limits and drop time less than 400 ms. An on-line system to monitor the drop time of control rod (CR) during scram was commissioned. Similarly a system was developed to measure friction force of CR during power operation. The 3 s interlock on CR raise movement, which was introduced before the first criticality was deleted as it was giving rise to large time in raising power and high start up duty demand on CRDM motors. The lower parts of two CRDM were replaced, one due to failure of translation bellows, and another due to failure of gripper bellows. Leaky silicone bellows of one CRDM was replaced in-situ.

During commissioning in 1985, when the sodium temperature were progressively raised to 350°C for isothermal tests, azhimuthal temperature difference (~ 80°C) in reactor vessel in the cover gas region was noticed. This resulted in tilting of reactor vessel and shift in grid plate as measured by displacement measuring device (DMD). This was investigated to be due to non-uniform natural convection currents in cover gas space [6], This was overcome by injecting helium to the argon cover gas to form a double layer above sodium level to suppress the rising convection currents.

The better heat transfer properties of helium also helped in homogenizing the circumferential temperature thus keeping grid plate shift well within limits.

2.4. Steam generator

The heat produced in reactor is transferred to tertiary circuit through steam generator. There are four identical modules of 12.5 MWt capacity located inside an insulated casing. It is a forced circulation, counter current, shell and tube once through type of steam generator.

Each SG module consists of seven tubes arranged in a triangular pitch inside the shell. Both shell and tube are made of niobium stabilized ferritic steel.

Sodium flows through the shell, which is not insulated to facilitate removal of decay heat by natural convection of air in the casing in case of off-site power failure/station blackout. SG is a critical equipment in fast reactors due to highly reactive nature of sodium when it comes in contact with feed water/steam in case of a SG tube leak. The tube volume is minimized to reduce the amount of water/steam available for reacting with sodium in case of tube failure. Provisions are made in the design to detect and take safety action such as shut down of reactor and isolation of steam generator in case of a tube leak at an early stage itself in order to avoid wastage.

Steam generator leak detection system (SGLDS) has been provided to detect small steam/water leak (few mg/s to few g/s) into sodium. This is a diffusion type meter wherein sodium sample from SG is passed through nickel tubes of 0.3 mm thick. Vacuum is maintained outside the tube with the help of sputter ion pump (SIP) and partial pressure of hydrogen is measured with the help of Quadra pole mass spectrometer (MSM) and SIP current.

Safety actions are initiated on detection of small leak. The system is calibrated by injecting hydrogen into SG.

The performance of MSM was not satisfactory due to failure of filament frequently and spurious trips due to noise pickup, etc. As the sputter ion pump current is a reliable signal, which is also proportional to leak rate, the logics were modified to take safety actions from sputter ion pump current in place of MSM signal.

Initial calibration of SGLDS at different sodium temperatures by injecting known quantity of hydrogen indicated that the response of the system is poor below 250°C. Hence it was decided to raise sodium temperature to 250°C from 180°C for admitting water into SG. The procedure for power raising was suitably modified and the system performed well. Subsequently a separate system, Hydrogen in Argon Detector (HAD), which can detect water/ steam leak at a sodium temperature of 180°C, was installed in expansion tank cover gas space and commissioned. In this system the cover gas argon is sent to a nickel tube and the presence of hydrogen outside the tube is detected by a Thermal Conductivity Detector (TCD). As majority of hydrogen produced is collected in the cover gas spaces of expansion tank, at lower temperatures, the system responds well at lower sodium temperature of 180°C. Performance of the system is being observed.

To protect against medium leaks of water/steam into SG (few gm/s to kg/s) the expansion tank pressure signal is used for initiating safety action such as shutdown of reactor and isolation of steam generator. Also a rupture disc is provided in the cover gas space of expansion tank to protect against over pressurization.

For protection against large water/steam leak into SG, rupture discs are provided on either side of SG. Rupture of rupture discs is detected by sodium spark plug detectors, which initiates safety actions as mentioned above. The reaction products are diverted to cyclone separator wherein the hydrogen gas produced is separated from the other reaction products and sent out to atmosphere through chimney.

To take care of sodium leak into the casing, collection trays with leak detectors are provided. Minute leak into the casing is detected by sodium aerosol detectors (SAD). Provisions are made to flood argon to SG casing to quench sodium fire in case of a leak.

The performance of the SG till now (12 500 h) has been satisfactory and there have been no incidents of tube failure or sodium leak into the casing.

3. INCIDENTS/UNUSUAL OCCURRENCES

3.1. Fuel handling incident

During an in-pile transfer operation in May 1987, a complex mechanical interaction occurred within the reactor vessel causing damage to the fuel handling (FH) gripper, the subassembly held by the gripper, guide tube and several reflector SA [2, 7]. The bent SA was forcibly extracted through the guide tube. The bent of guide tube was estimated by three techniques, viz., optical inspection, ultrasonic air gauging and mechanical disc gauging. The guide tube was cut and removed in two pieces using specially designed tools. The incident was investigated in detail and found to be due to system deficiencies combined with human error. Modifications viz., mechanical stopper for fuel-handling gripper and redundant interlocks for plug rotation authorization were implemented. Also proper maintenance and operating procedures for FH mechanisms were evolved and reviewed by an expert committee. It took two years to recover from this incident and the reactor was restarted in May 1989. After incorporating these modifications, 185 charging/discharging and 177 transfer operations in 10 fuel-handling campaigns have been successfully carried out over the past 13 years without any incident.

3.2. Malfunctioning of core cover plate mechanism (CCPM)

The outlet temperature of 84 core SA is monitored by thermo couples which are housed in CCPM. The fuel SA thermo couple signals are scanned by computer to generate trip signals.

During normalization of pile after fuel handling operation in July 1995, CCPM could not be lowered to normal working position from fuel handling position. Various operations resulted in its getting stuck at 81 mm position above top of SA heads. The likely causes were attributed to mechanical obstruction at the top, below the core cover plate or within the mechanism. Based on systematic investigations viz., checking of obstruction, scanning the space below the core cover plate and above the SA head by ultrasonic under sodium scanner, ensuring leak tightness of bottom metallic bellows etc., it was confirmed that the sticking is in the inter seal space between command tube and outer sheath. A safe jacking down force of 780 kg was applied and CCPM was brought to normal position and made functional. However, during the next fuel handling operation in July 1996, CCPM again got stuck up at 80 mm position and it could not be normalized even after repeated trials. The exact cause for malfunctioning of CCPM could not be identified [2, 8].

Experiments were carried out on power to measure the fuel SA outlet temperature with CCPM stuck at 80 mm position and a temperature attenuation of 7% (average) was found in Mark I SA. However, this attenuation is large for Mark II SA where sodium flow is less. Studies were conducted to find out the probability of plugging during reactor operation and found to be acceptable. 3D analysis of outlet plenum thermal hydraulic was carried out to establish the level of plugging that can be detected viz., allowable plugging for fuel clad integrity.

The studies indicated that a flow reduction to 60% of the nominal flow for the Mark II SA in the core periphery could be detected with CCPM at 80 mm position, whereas a flow reduction to 45% is required to cross the clad hot spot temperature. The design provisions such as radial entry of sodium flow into the SA and high purity of sodium maintained, rules out blockage of flow through SA of the order mentioned above.

PSA studies were carried out based on available data from various fast reactors to establish the probability of plugging in SA. Based on these studies, lowering of scram thresholds for core ΔT and core mean temperature (θm) from the fuel SA thermocouple was done and reactor operation continued with CCPM at 80 mm position.

An eddy current flow meter was developed for measuring actual flow through the FSA during shut down by installing it in FH guide tube and out of pile tests were carried out. This is planned to be used in the reactor to estimate actual flow for selected SA. Also, out of pile mockup trials were carried out for rectification of stuck CCPM.

3.3. Na/Nak leak incidents

While preheating of secondary cold trap during initial commissioning, about 2.5 litre of Nak leaked out from the Nak jacket through the spark plug type level probe. Investigation revealed failure of level probe due to pressure build up during preheating due to non-availability of adequate expansion space. Modifications viz., capping of level probe, providing an argon pot to allow expansion was carried out to prevent recurrence.

During routine sampling of secondary sodium system, about one litre of sodium leaked out through the swage lock coupling of the flow through sampler. To prevent recurrence, provision was made in the sampler for helium leak testing prior to valving in sodium.

During maintenance on the pressure regulator in the argon supply system, about 2 litre of NaK backed up from the NaK bubbler provided for argon purification and leaked out. As a remedial measure a back flow trap was introduced in the circuit.

3.4. Water leak for SG sub-headers

In January 1993, when SG was put in service for the first time, after 70 hours of operation, a water leak took place due to a linear pinhole defect in the end cap of one of the orifice assemblies at SG inlet. All similar end caps were ultrasonically inspected and four more were found to have similar defects. The leaking cap was replaced and additional covers were welded for defective caps. The failure was attributed to material defect.

All the four modules of SG are provided with flanged orifices in the water sub headers for flow measurement to study SG h stability. In August 1993 when reactor was operating at 9 MWt, feed water was found to be leaking through the orifice flanges. Investigations revealed that leak tight orifice flanges under ambient conditions tend to develop leak under operating conditions due to differential expansion between water sub headers and SG modules. All the orifice flanges were replaced with welded spools with integral orifices.

In February 1998, while readjusting the setting of SG safety valves in cold state, water leak was observed in one of the bosses in experimental thermo well in the steam sub header of one of the SG modules. The leaking boss and plug were replaced with dummy piece and all other similar welds were checked for any defects by liquid penetrant inspection. Investigation revealed that failure is due to lack of heat treatment during fabrication.

3.5. Seizure of main boiler feed pump [9]

In April 1992, while preheating feed water system, abnormal noise was heard from the pump and the pump got seized.

Investigation revealed that the failure was due to cavitation. The net positive suction head (NPSH) available to the pump was found to be very close to the required NPSH and it further reduced during operating transient. Modifications to improve NPSH available were carried out viz., rerouting of balancing leak off line water back to suction tank instead of pump suction to reduce suction temperature, continuous cold water injection to the suction, additional re-circulation line to avoid pump operating at low flows. Also steam heating of deaerator was done using steam from package boiler. This resulted in a delay 8 months to put SG in service.

In May 2001, while starting one of the main boiler feed pump (MBFP) in cold condition, abnormal noise was heard and the pump got seized. Damages to the pump noticed are similar to above incident. Investigation is being carried out to find out the cause.

3.6. Reactivity transients

In November 1994, when reactor was operating at 10.1 MWt, reactor power started increasing without any movement of control rods. Control rods were lowered to restore the power to 10.1 MWt. However, the power continued to rise and reached 10.45 MWt within a minute. Control rods were lowered by 7.6 mm (22 pcm) to bring back the power of 10.1 MWt. The reactivity recorder registered a spike of 3 pcm during the incident. No permanent gain in the reactivity was observed before and after the incident.

In April 1995, when reactor power was stabilized at 7.1 MWt in the process of power raising to 10 MWt, there was a sharp increase in power of around 450 kWt in 7 s and reactor underwent scram on high positive reactivity (treshold + 10 pcm). The reactivity recorder also indicated a spike of 10 pcm. A permanent gain of about 13.8 pcm was observed before and after the incident.

Both these incidents were analyzed by a task force constituted. Totally 19 postulates were studied to find out possible cause for the transients. They were related to change in process parameters, movement of absorber rod, movement of fuel, reactivity change due to sodium void and moderator ingress to the core. Several of these were tested with reactor operating at low power as well as high power. However, the cause of the incidents could not be identified. Ingress of sodium oil reaction products (due to leak from sodium pump) into the core was considered as one of thee probable cause. Sub-critical operation at 400°C for one week did not reveal recurrence of the incident. Since the incident has occurred during high power operation, it was decided to operate the reactor after increasing reactivity scram treshold. A fast recording data acquisition system was developed to gather more data if the incident recurs.

In 1998, when reactor was being operated at 8 MWt for irradiation of zirconium-niobium (Zr-Nb) alloy, reactivity transient, that is repeatable in nature, was encountered. Following are the observations:

1) The transient is self-limiting and the power remains stable after the transient.

2) The reactivity gained (~ 30 to 40 pcm) while raise in power is lost while manually lowering the power. There is no permanent gain in reactivity. The power increase is around 700 to 800 kWt.

3) The on set of transient occurs at a specific value of mean core gradient temperature ($\Delta\theta m$) of around 90 to 100°C across the core for a given primary sodium flow. The transient can be observed at any power level, if the above $\Delta\theta m$ is reached.

4) The value of $\Delta\theta m$ at the onset of the reactivity transient increases with increase in primary sodium flow and the magnitude of the transient also comes down.

5) At a flow of equal to more than 460 m^3/h the transient does not occur.

The cause of the incident was investigated in detail by the task force. Following tests were carried out:

- Suspecting the experimental Zr-Nb SA could be the cause for the incident, these were removed from the core and experiment repeated. The transient was found to be recurring.

- The isothermal temperature flow and power coefficient were measured and no abnormality was observed.

- To test the effect of control rod for the transient, experiments were conducted with one control rod at higher position (405 mm) while power is maintained by other 5 control rods and 5 control rods at 405 mm while power is maintained by 6th control rod. In both the tests transient were found to be recurring.

- In order to establish the self-limiting nature of the transient and to ensure that there is no second transient after the first one, testes were conducted. Reactor was operated beyond the transient region for more than 30 m and the power was found to be stable. While lowering the power the negative transient of similar magnitude took place and the power was stable thereafter.

From the experiments conducted, it is concluded that the transient is due to slight, thermally induced geometric changes in the core, which happens at low flow rates. It has come to light because the reactor was operated at low power and low flow rates, for Zr-Nb irradiation, in the regime it manifests. Since reactor is normally operated at higher power (12 to 15 MWt) and correspondingly higher flow rates (> 450 m^3/h), there is no transient seen. With this, clearance was obtained from safety authorities for continuing operation of reactor at high power after establishing stable regime for power operation whenever core configuration and operating power is changed.

3.7. Water leak from BSC coils inside biological shield concrete

The reactor vessel of FBTR is surrounded by a safety steel vessel and further by two types of concrete namely 600 mm thick biological shield and 900 mm thick structural concrete.

A gap of 30 mm is provided between the two concrete to take care of differential thermal expansion. The biological shield concrete is cooled by circulating water through 180 coils embedded in concrete. The biological shield cooling (BSC) system has two distribution headers, each have 100% capacity. Each header has six sub-headers with individual isolation valves.

Each sub-header cools 60° sector of the concrete. The coils from the sub headers are laid in the concrete in such a way that two adjacent coils are from different headers so that even if one sub-header is not available, concrete cooling is not affected.

In August 2000, high inventory loss from the BSC system was observed. Also, water seepage was observed from the structural concrete manhole cover. When one of the sub-headers in header-A (A3), which cools 60° sector in the southwest portion of the concrete was isolated, the leak stopped. The leaky sub-header was isolated, water collected inside A1 cell (gap between steel vessel and biological concrete) was drained and reactor operation continued. There was no increase in biological shield concrete temperature.

In May 2001, again, inventory loss in BSC system and water seepage was observed. Reactor, which was operating at low power, was shutdown and investigation was carried out. When the sub-header in header-B (B5), which again cools the same 60° sector in the southwest portion of the concrete, was isolated, the leak stopped. Detailed investigation revealed existence of minute leak in two more sub-headers in header B (B1 and B4).

Isothermal tests conducted at higher sodium temperature of 375°C indicated that it is not possible to maintain the concrete temperature within limits as the first two leaky sub headers (A3 & B5) cools the same l/6th region of the concrete. Hence reactor operation at high power was suspended.

Injecting proprietary formulation sealants arrested the leak points in the coils of the two sub headers and the coils were tested for healthiness. For remaining coils, a global sealant treatment was carried out to arrest micro leaks and the system was normalized and power operation resumed.

As a future measure, four holes were drilled in the structural concrete up to the gap between biological and structural concrete to drain out water. This is to prevent entry of water to Al cell in case any leak in coils recurs in future and also provide safe draining passage for the leaking water collecting in the interspace between structural concrete and biological concrete. Investigation carried out indicated that the leaking point is the socket weld portion of the coil located in the inaccessible region and it could have happened due to crevice corrosion.

4. DECONTAMINATION EXPERIENCE

Various components and mechanisms which are working in active sodium of the primary circuit need to be decontaminated before they are inspected, sent for maintenance or dismantled for repair without the risk of sodium fire and radio activity. The decontamination facility consists of three pits. Pit No.1 & 2 are used for large components viz., IHX, pump, etc. Pit No.3 is used for small components like guide tube, CRDM, level probes and core co-ordination measuring device (CCMD).

Decontamination of equipments is done by admitting steam from an electrode package boiler. For large components like IHX and pump, the bulk sodium deposited on the equipment is removed by heating with the help of surface heaters prior to steam admission. The drained sodium is collected in collection trays and disposed off. During steam admission hydrogen concentration and pressure build up in the pit is constantly monitored. Rupture discs are provided in the pit to protect against high-pressure build up. In case the H_2 concentration increases to > 2% and/or build up pressure, steam admission is stopped and the pit is flushed with argon. The effluent gases are sent to stack. When the hydrogen concentration reduced to < 1%, the steam admission is restarted. The procedure is repeated till there is no increase in H_2 concentration and pressure build up indicating completion of decontamination. The equipment is now sprayed with water and the pit is filled with water to remove any sodium sticking on the equipment. The liquid effluent is drained and pumped into liquid effluent tanks for disposal.

The primary sodium pumps were initially fitted with strainers in place of non-return valves during first criticality to trap any construction debris/impurities in sodium. Pumps were removed from primary system, decontaminated, refitted with NRVs and put back in the circuit. One CRDM, which was removed from reactor due to, failure of metallic bellows was decontaminated and its non-active parts were retrieved for reuse. Similarly guide tube under sodium scanner, CCMD and periscope were decontaminated after their use in reactor.

The fingers of charging and discharging flasks were periodically decontaminated in an alcohol pot located in RCB whenever difficulty is noticed in operating gripper fingers due to sodium sticking. The grippers are cooled to atmospheric temperature prior to lowering in alcohol pot. During decontamination of fingers in alcohol pot, pressure build up in pot is constantly monitored.

Rupture disc provided for decontamination pit failed once due to aging and degradation and was replaced. Generally the decontamination experience in the plant has been satisfactory.

5. MODIFICATION IN REACTOR PROTECTION SYSTEM

During initial periods, there were large number of spurious scram due to electro magnetic noise pickup by start up and delayed neutron detector channels and due to “cross talk” of fine impure test (FIT) pulse. Remedial measures viz., separation of signal cables from power cables, improvements in grounding system and reshaping FIT pulse were carried out and system performance improved.

A set of pre-start up channels with high sensitivity BF_3 proportional counters was commissioned. These channels along with startup channels help in smooth start up of the reactor with inherent neutron source, thus eliminating the need for high strength auxiliary neutron source.

During reactor operation SG got isolated on waterside due to disturbance in control power supply making heat sink not available. Reactor did not trip on low feed water flow due to zero error in flow transmitter and reactor was shutdown manually. In order to prevent recurrence of incident, protection circuit was modified to trip the reactor in case of isolation of SG directly and threshold setting of low feed water flow was increased to 50% of nominal flow.

Since the number of LOR and scram during high power operations was very high, an exercise was carried out to optimize the trip parameters. As a first step, high winding temperature trips of sodium pump drive system motors (88 No) were deleted and control panels of these systems were housed in an air conditional atmosphere. These steps have vastly improved performance of the drive system. As second step the trip parameters of reactor protection system were reviewed and the following modifications were carried out to improve the reliability of the system without compromising the safety.

- The positive reactivity scram is manually inhibited during power raising and made effective during steady power operation.

- Log P scram set at 110% of nominal power was giving spurious alarm due to inadequate margin between the nominal power and alarm threshold. Hence its setting was jacked up to 125%. In order to ensure that Lin P channel is available, a new log P scram at 10% of nominal power was introduced which will be inhibited at low power (3% of nominal power) by Lin P signal.

- LOR on low current in electro magnetic (EM) coil of any CRDM was deleted. In case of dropping of any control rod, reactor will be shutdown either by -ve reactivity or control rod level discordance.

- All class II LOR (power setback) parameters have been changed to class I LOR parameters.

- LOR threshold of control rod level discordance was increased from +/- 20 to +/- 40 mm to provide more operating margin.

- Reactor trips from computer based central data processing system (CDPS), which do not involve computation or processing, were shifted to hardwired circuit to improve reliability of CDPS.

All the above modification improved the reliability of reactor protection system and spurious trips reduced.

6. RADIATION PROTECTION

The activity release and man-rem expenditure in fast reactors are generally expected to be very low compared to thermal reactors. In FBTR the total stack release and man-rem expenditures are 11.545 TBq and 16.195 man-rem respectively from criticality till date, which is much less than the allowed limits. The total stack release and man-rem expenditures are 1318.6 GBq and 805 man-mille rem respectively for the year 2001.

7. CONCLUSION

FBTR has been fully commissioned with small core up to a power level of 13.4 MWt and the performance of all the safety related systems has been satisfactory. Large number of modifications was carried out based on experience feed back and analysis of various incidents to improve system performance. Construction, commissioning and operation of FBTR have given considerable amount of experience and confidence, which will help in its smooth and sustained operation at nominal power and will also give useful feedback for the design and commissioning of large fast reactors.

REFERENCES

[1] RAMALINGAM, P.V., et. al., Operating experience of fast breeder test reactor, Proc. 7th International Conference on Nuclear Engineering, 19-23 April 999, Tokyo, ICONE-7337 (1999).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Unusual Occurrences in Fast Breeder Test Reactor, IAEA-TECDOC-1180, Vienna (1999).

[3] GANGULY, C., et. al., Development and fabrication of 70% PuC-30% UC fuel for fast breeder test reactor in India, Nuclear Technology, Vol. 72 (1986) 59.

[4] BALDEV RAJ, Post irradiation examination of mixed (Pu, U)C fuels irradiated in fast breeder reactor, IAEA IWG-FR Technical Committee Meeting on Influence of High Dose Radiation on Advanced Reactor Core, Structural and Fuel Material (1997).

[5] RAJENDRAN, B., Experience gained during commissioning, Proc. 4th Intl Conf Liquid Metal Engineering and Technology (LIMET 88), organized by Societe Francaise d' Energie Nucleaire, 17-21 October 1988, Avignon, France (1988) 705.

[6] SRINIVASAN, G. et al., Thermo-mechanical behaviour of FBTR reactor vessel due to natural convection in cover gas space, Proc. 4th Intl Conf Liquid Metal Engineering and Technology (LIMET 88), organized by Societe Francaise d' Energie Nucleaire 17-21 October 1988, Avignon, France (1988).

[7] BHOJE, S.B., et al., Design and construction of fast breeder test reactor, paper presented in the IAEA International Symposium on Fast Breeder Reactor, Experience and Future Lyon, France, 1985.

[8] BHOJE, S.B., et al., Status of fast reactor development in India, paper presented in the 29th Annual Meeting of the International Working Group on Fast Reactors, Aktau, Kazakhstan, 1996.

[9] KALE, R.D., et al., Analysis of failure of axial trust bearing and thrust bearing of a small capacity boiler feed pump for a nuclear power plant, paper presented in the 4th Asian International Conference on Fluid Machinery, 1993.

OPERATIONAL EXPERIENCE AND UPGRADING PROGRAM OF THE EXPERIMENTAL FAST REACTOR JOYO

T. AOYAMA, T. ODO, S. SUZUKI, S. YOGO
Japan Nuclear Cycle Development Institute (JNC), Ibaraki, Japan

Abstract

Twenty years of successful operations at the experimental fast reactor JOYO provide a wealth of experience covering core management, chemical analysis of sodium and cover gas for impurity control, natural convection tests, upgrade of fuel failure detection system, corrosion product measurement, development of operation and maintenance support system, and replacement of major components in the cooling systems. Some of the data obtained is stored in a database to preserve the related knowledge. This experience and accumulated data will be useful for the design of future fast reactors.

1. INTRODUCTION

The experimental fast reactor JOYO at the Japan Nuclear Cycle Development Institute's Oarai Engineering Center attained initial criticality in April 1977 and was 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. Thirty-five duty cycle operations and thirteen special tests with the MK-II core were completed by June 2000 without any fuel pin failures or serious plant trouble. The reactor is currently being upgraded to the MK-III core. This paper provides a review of the operational experiences obtained through the JOYO's operation.

2. SPECIFICATIONS, PLANT DESCRIPTION AND OPERATION HISTORY OF JOYO

JOYO is a sodium cooled fast reactor with mixed oxide (MOX) fuel. The main reactor parameters of the MK-II irradiation bed core are shown in Table 1, which compares the MK-II with the future MK-III core.

TABLE 1. MAIN CORE PARAMETERS OF JOYO

Items			MK-II	MK-III
Reactor Thermal Output		(MWt)	100	140
Max. No. of Test Irradiation S/A			9	21
Core Diameter		(cm)	73	80
Core Height		(cm)	55	50
^{235}U Enrichment		(wt%)	12(J1)/18(J2)	18
Pu Content		(wt%)	≦ 30	≦ 30
Pu fissile Content (Inner/Outer Core)		(wt%)	~20	~16/21
Neutron Flux	Total	(n/cm^2/s)	4.9×10^{15}(J2)	5.7×10^{15}
	Fast(>0.1MeV)	(n/cm^2/s)	3.2×10^{15}(J2)	4.0×10^{15}
Primary Coolant	Temp. (Inlet/Outlet)	(℃)	370/500	350/500
Operation Period		(days/cycle)	45(J1)/70(J2)	60
Reflector/Shielding			SUS/SUS	SUS/B_4C
Max. Excess Reactivity (at 100 ℃)		%Δk/kk'	5.5	4.5
Control Rod Worth		%Δk/kk'	≧ 9	≧ 7.6

The MK-II driver fuel plutonium content is about 30wt%. Initially the ^{235}U enrichment was about 12wt%, however this was increased to 18wt% in 1987 to provide enough excess reactivity so that the core burnup was increased. Consequently, the operational period was extended from 45 to 70 days and the plant availability increased. Since 1998, some MK-III driver fuel subassemblies, which have the same specification as the MK-II except a shorter fuel stack length, were loaded in the outer region of the core. Figure 1 shows an example of the MK-II core configuration.

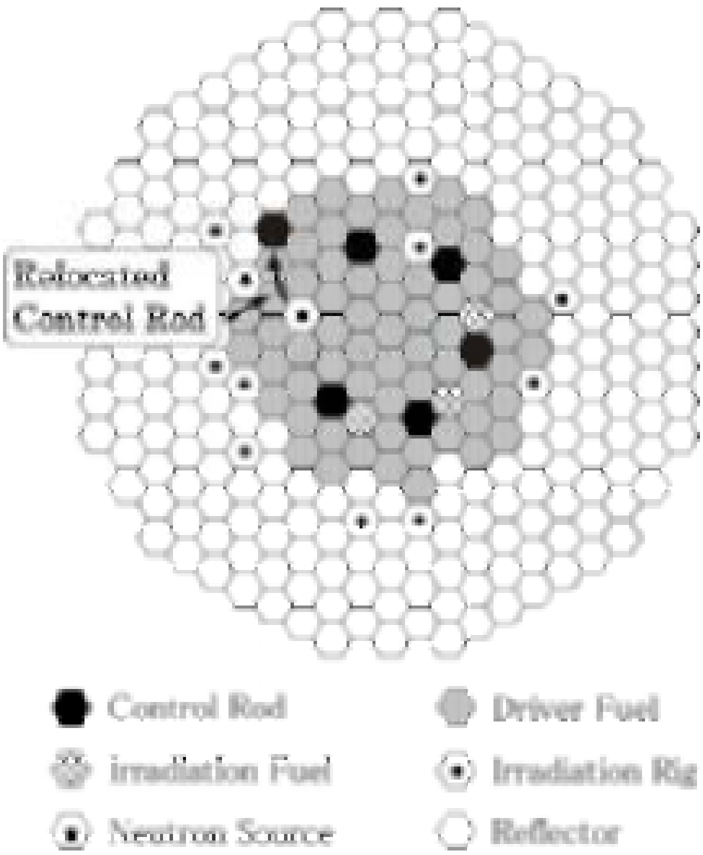


FIG. 1. JOYO MK-II core configuration.

Six-control rod subassemblies made of 90% enriched B_4C were used in JOYO MK-II and were located symmetrically in the third row. In 1994, one control rod was moved to the fifth row to provide a position for irradiation test assemblies with on-line instrumentation. Since then, the control rod subassemblies have been loaded asymmetrically. The JOYO cooling system has two primary sodium loops, two secondary loops and an auxiliary cooling system. The cooling system uses approximately 200 tons of sodium. In the MK-II core, sodium enters the core at 370°C at a flow rate of 1 100 tons/h/loop and exits the reactor vessel at 500°C. The maximum outlet temperature of a fuel subassembly is about 570°C. An intermediate heat exchanger (IHX) separates radioactive sodium in the primary system from non-radioactive sodium in the secondary system. The secondary sodium loops transport the reactor heat from the IHX to the air-cooled dump heat exchanger (DHX). A flow diagram of the cooling system of MK-II core is shown in Fig. 2.

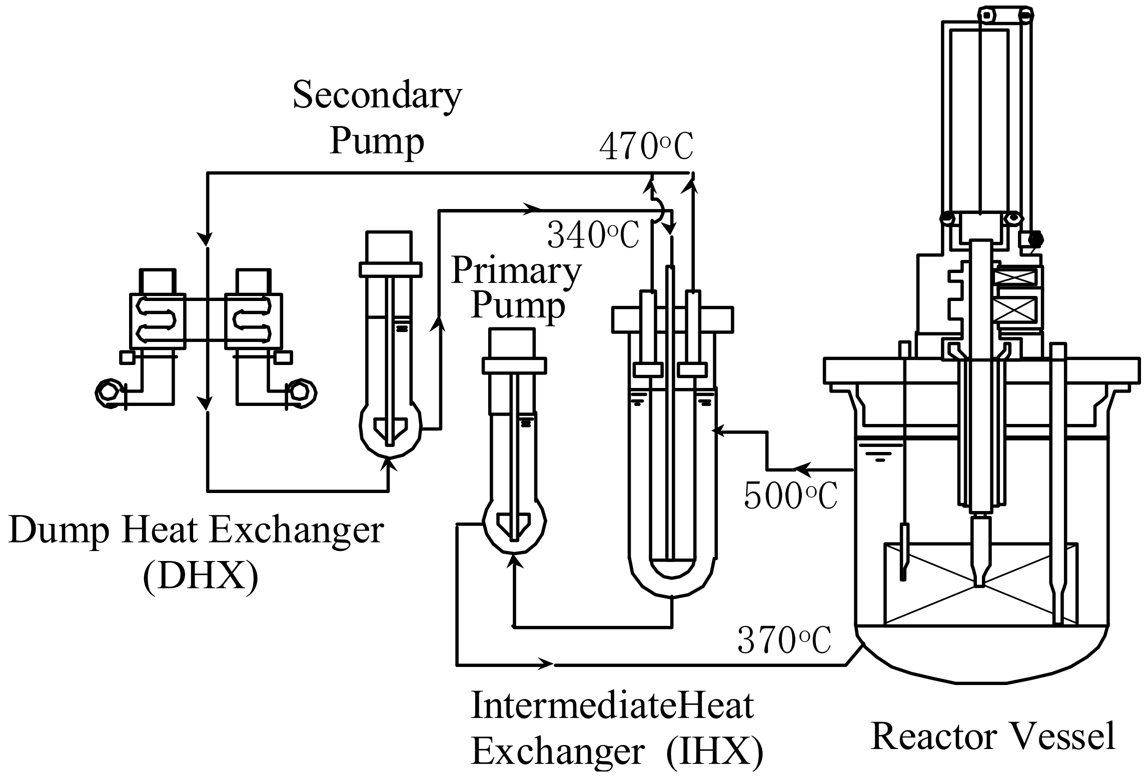


FIG. 2. JOYO cooling system diagram.

The operating data and history of the JOYO MK-II core are shown in Table 2 and Fig. 3. The reactor operated for 48 000 hours and the integrated power generated was 4 400GWh. During the MK-II operation, 382 driver fuel subassemblies and approximately 47 000 fuel pins were irradiated. A peak burnup value of 86.0 GWd/t was attained for the MK-II driver fuel without any fuel pin failures.

TABLE 2. JOYO MK-II OPERATING DATA

Operation Time(Accumulated)	48,000 hrs
Heat Generation(Accumulated)	4,400 GWh
Max. Fuel Burn-up	
Driver Fuel	86 GWd/t
Irrad. Fuel	142 GWd/t
No. of Irradiated Fuel Subassemblies	382

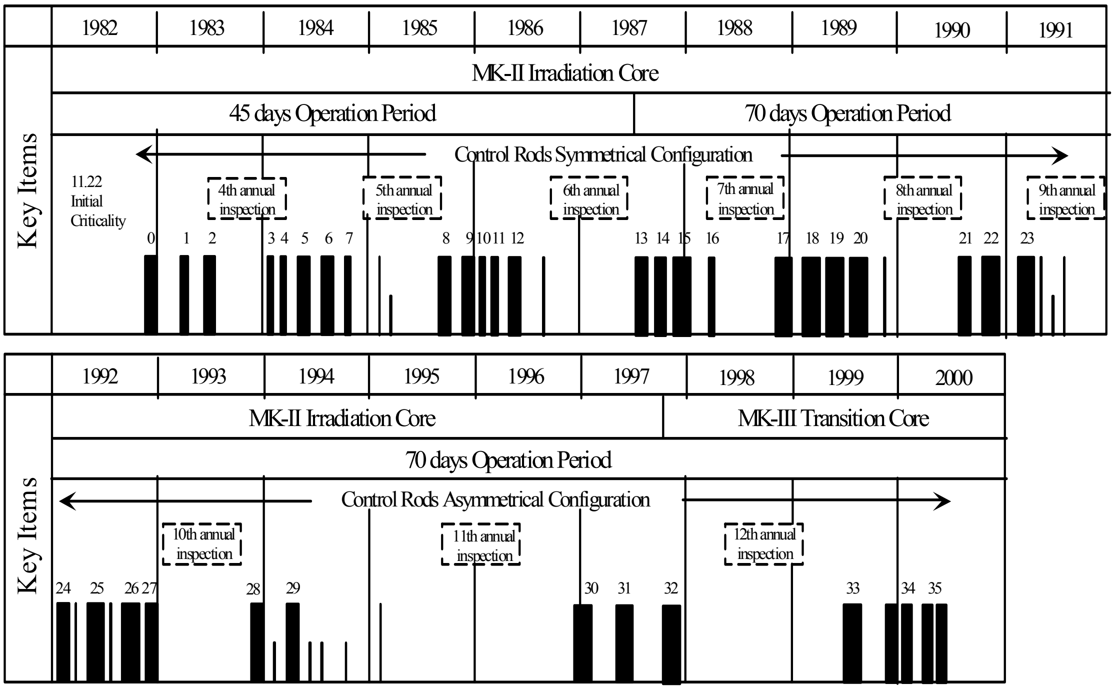


FIG. 3. JOYO MK-II core operation history.

3. CORE MANAGEMENT EXPERIENCE

A core management code system has been developed to predict the core parameters for operation and refueling plans within the design limitations. The nuclear calculation is based on diffusion theory and corrected with a bias method. Results from core physics tests and Post Irradiation Examinations (PIE) have been used to confirm the accuracy of these predictions. These verifications are also important to conduct various irradiation tests accurately. This section describes the method and verification for core and fuel management used with the JOYO MK-II core.

3.1. Method

The MAGI calculation code system [1, 2] was developed to predict the reactor parameters required for core and fuel management of the JOYO MK-II core.. MAGI is a neutronic and thermo-hydraulic coupling code system that calculates the excess reactivity, power distribution, fuel burnup, coolant flow rate and temperature condition of each subassembly. MAGI uses diffusion theory with seven neutron energy groups for the nuclear calculations. The neutron cross section was collapsed from the 70 group JFS-3-J2 cross section set [3] processed from the JENDL-2 library. It was updated to the JFS-3-J3.2 cross section set based on the JENDL-3.2 library [4] in 2001. The gamma-ray cross section that includes delayed fission gamma-ray was processed from the JENDL-2 library to improve the calculations of gamma heating in stainless steel. The core configuration was modeled in three-dimensional Hex-Z geometry for each operational cycle. The actual reactor power history was used in the burnup calculation. Figure 4 shows the MAGI system outline.

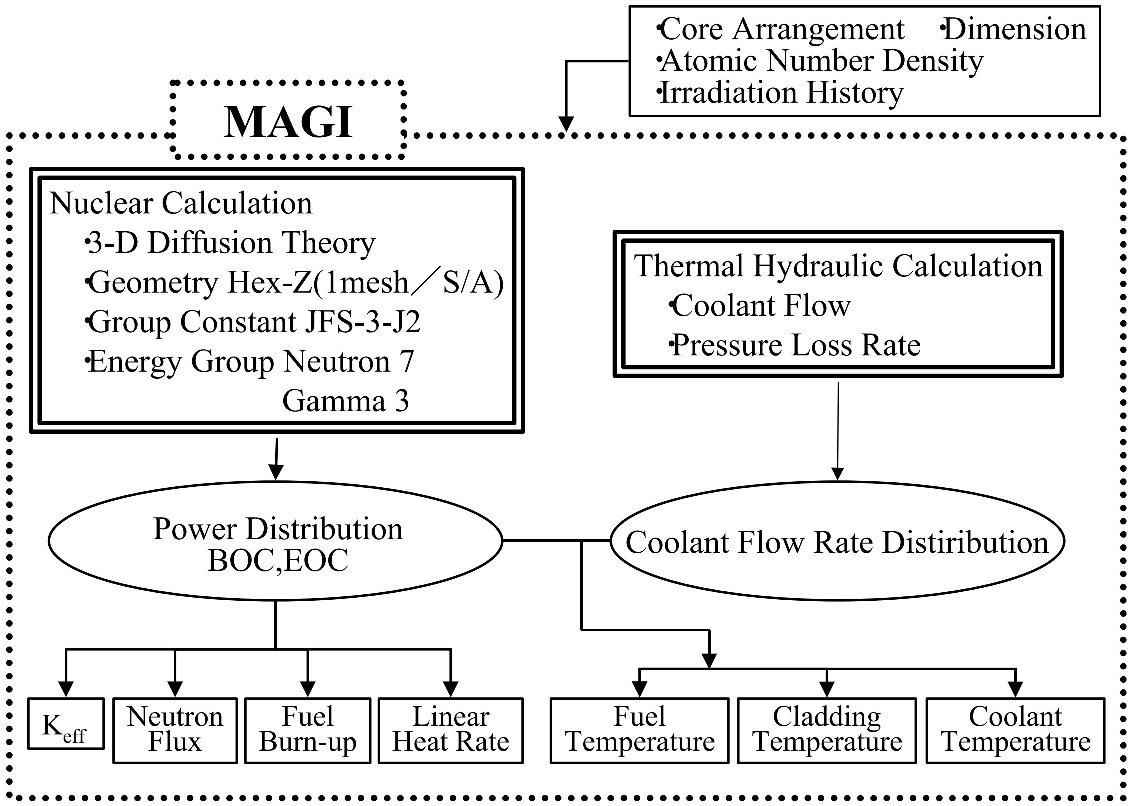


FIG. 4. JOYO MK-II core management code system.

MAGI supported the refueling plan for each operational cycle. The number of fresh driver fuel subassemblies and their position were determined within design limitations of neutronic and thermal parameters with consideration for the following items:

(1) Burnup of fuel subassembly;
(2) Operation period of the cycle;
(3) Neutron fluence and temperature condition of the irradiation test subassemblies.

The number of discharged spent fuel subassemblies ranges from 10 to 12, about one sixth of total driver fuel subassemblies. The frequent refueling affects the core characteristics and irradiation conditions significantly so the MAGI calculation needs to be verified in each operation cycle.

3.2. Verification of core management

3.2.1. Core physics tests in each operational cycle

3.2.1.1. Criticality and burnup reactivity

The excess reactivity in a zero power critical condition was measured at the reactor start-up of each operational cycle. Measured data were compared with the MAGI calculated values that included the bias factor (C-E) correction obtained for the previous cycle. The comparison of calculated and measured values is shown in Table 3.

TABLE 3. C/E VALUES OF CORE CHARACTERISTICS

Core Characteristic		C/E
Excess Reactivity	(%Δk/kk')	0.99±0.03
Control Rod Worth		(±0.1%Δk/kk')
Symmetrical	(%Δk/kk'/total)	0.99±0.02
Asymmetrical	(%Δk/kk'/total)	0.98±0.01
Burn-up Reactivity		
Coefficient	(%Δk/kk'/MWd/t)	0.96±0.05

By using this bias method, it was found that the excess reactivity after refueling can be well predicted within an error of 0.1%Δk/kk'. The burnup reactivity was determined by measuring the reactivity change during rated power operation. Measured values were compared with the MAGI burnup calculation and both agreed within 5% as shown in Table 3.

It is considered that the decrease of atomic number densities of major fissile nuclides as ^{235}U and ^{239}Pu are the dominant factor of burnup reactivity because of JOYO's small core size, which results in hard neutron spectrum and small internal conversion ratio (~0.3). Therefore, the burnup reactivity can be predicted accurately even at a high burnup.

3.2.1.2. Control rod worth

Every control rod's worth was calibrated with either the positive period method or the inverse kinetics method during the low power test of each operational cycle. The calculation was conducted with the CITATION code based on three dimensional diffusion theory using the same cross section, geometry and atomic number densities as MAGI. After correcting with the previous cycle's C/E, the calculated values were compared with the measurements.

Table 3 shows that both agreed within 2%. When one control rod was moved from the third row to the fifth row in 1992, the worth of the relocated rod was reduced to one third. The calculation accuracy was not changed significantly.

3.2.1.3. Reactivity coefficients

Reactivity coefficients are important for safety reasons and to account for the reactivity change associated with temperature and power changes during reactor operation. The measured values are used to predict the reactivity change for the next operational cycle.

The isothermal temperature coefficients were measured by taking the difference of reactivity at approximately 250 and 370°C during reactor start-up. The measured isothermal temperature coefficients were constant through the MK-II operation because they were determined mainly by radial expansion of the core support plate, which is independent of burnup. However, when the core region was gradually extended from the 32nd cycle, the isothermal temperature coefficients were decreased as predicted with the mechanism. Table 4 shows these values.

TABLE 4. MEASURED ISOTHERMAL TEMPERATURE COEFFICIENTS

Cycle	Number of Fuel S/A	Isothermal Temperature Coefficient ($\times 10^{-3}$% Δ k/kk'/℃)
MK-II Average	67	-3.98± 0.12
32	69	-3.67
33	71	-3.65
34	75	-3.47
35	76	-3.49

The power coefficients were measured at the reactor start-up and shutdown in each operational cycle. Figure 5 shows that the measured power coefficients decreased with increasing core burnup.

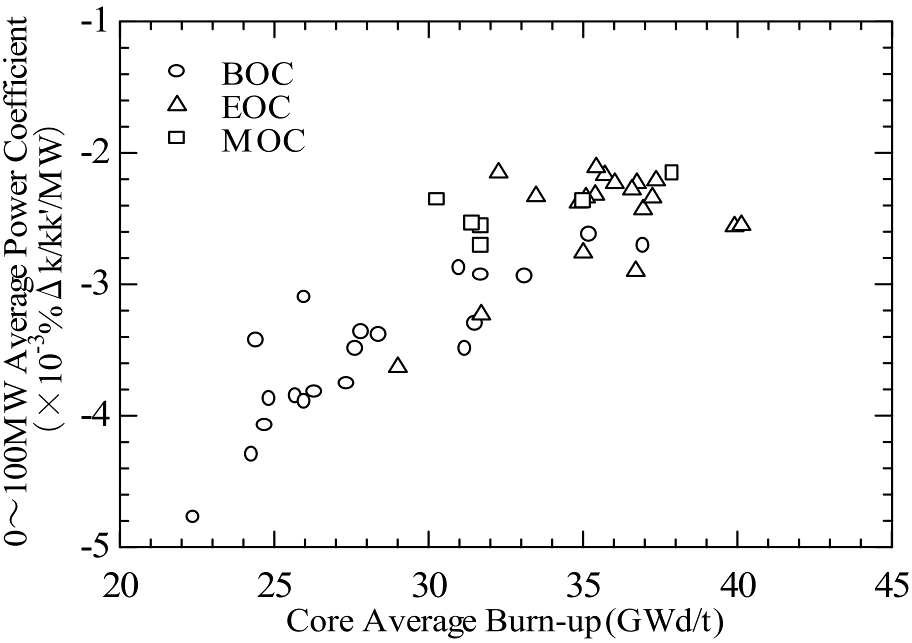


FIG. 5. Burnup dependence of power coefficient.

It was considered that the fuel thermal expansion, which is the major component of the power coefficient of JOYO, decreases at high burnup due to fuel restructuring during irradiation. It was also observed that the power coefficients varied depending on the reactor power as shown in Fig. 6. This phenomenon appeared to be due to a combination of the core bowing effect, fuel thermal expansion and Doppler effects. These causes need further investigation.

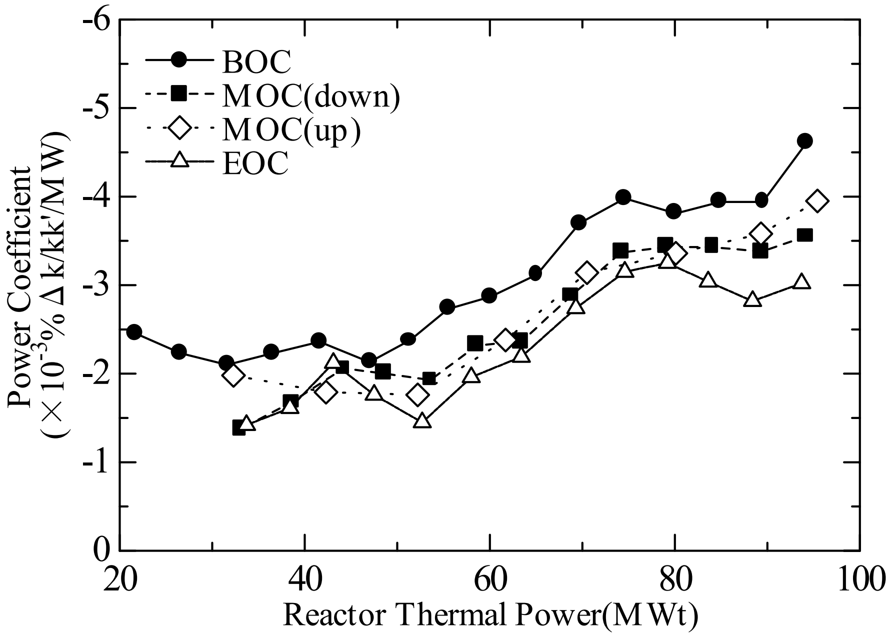


FIG. 6. Power dependence of power coefficient.

3.2.2. Post Irradiation Examination (PIE)

3.2.2.1. On-site burnup distribution measurement

The calculation accuracy of MAGI's axial power distribution was confirmed by comparing the relative distribution of measured (^{144}Pr) and calculated fuel burnup. The burnup measurement was taken in the JOYO spent fuel storage pond [5]. The spent fuel subassembly was contained in a stainless steel can, and it was set on a fuel-scanning device. Axial and circumferential gamma-ray distributions were measured using a high purity germanium (Ge) detector by moving the fuel scanning system vertically and by rotating the subassembly around the fixed detector.

As illustrated in Fig. 7, the MK-II spent fuel subassembly with a burnup of 62.5 GWd/t and cooling time of 5.2 years was measured. It was shown that the measured and MAGI calculated values were close except at the upper region of the fuel column. This difference was apparently due to the calculation error of neutron absorption by control rod.

3.2.2.2. Burnup ratio measurement by ^{148}Nd method

Chemical analysis of ^{148}Nd was conducted at the hot cell facility. As ^{148}Nd is one of the stable fission products and its fission yield is highly reliable, ^{148}Nd production obtained by destructive examination has been commonly used as a burnup index [6]. The calculated and measured burnup ratios for the MK-II spent fuel from 0.3 to 8.7 atom% is shown in Fig. 8. Measured results agreed with the MAGI burnup calculation within 5% in each row.

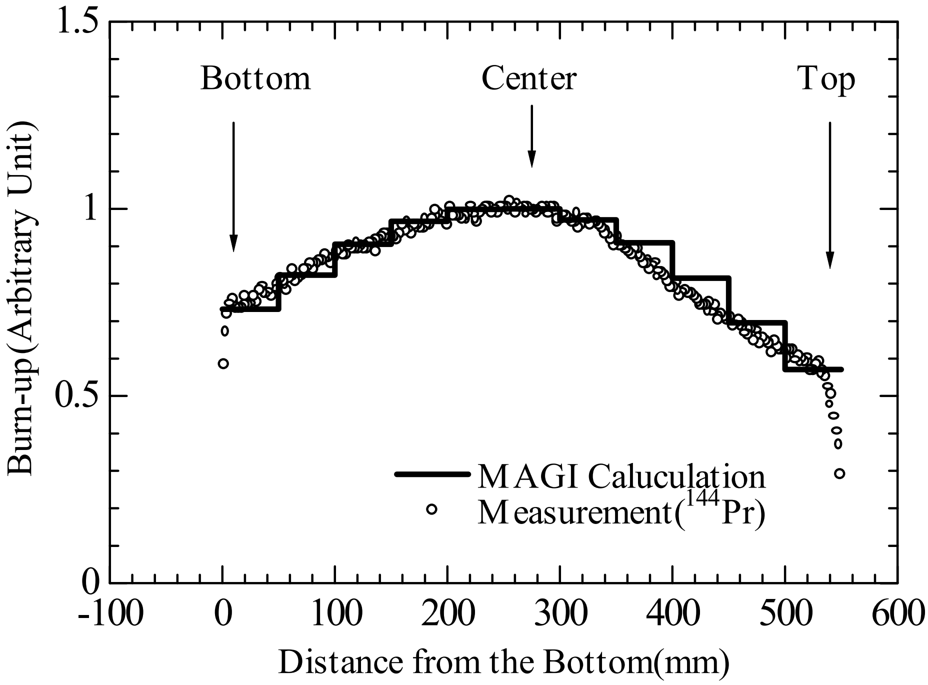


FIG. 7. Axial burnup distribution of JOYO MK-II driver fuel.

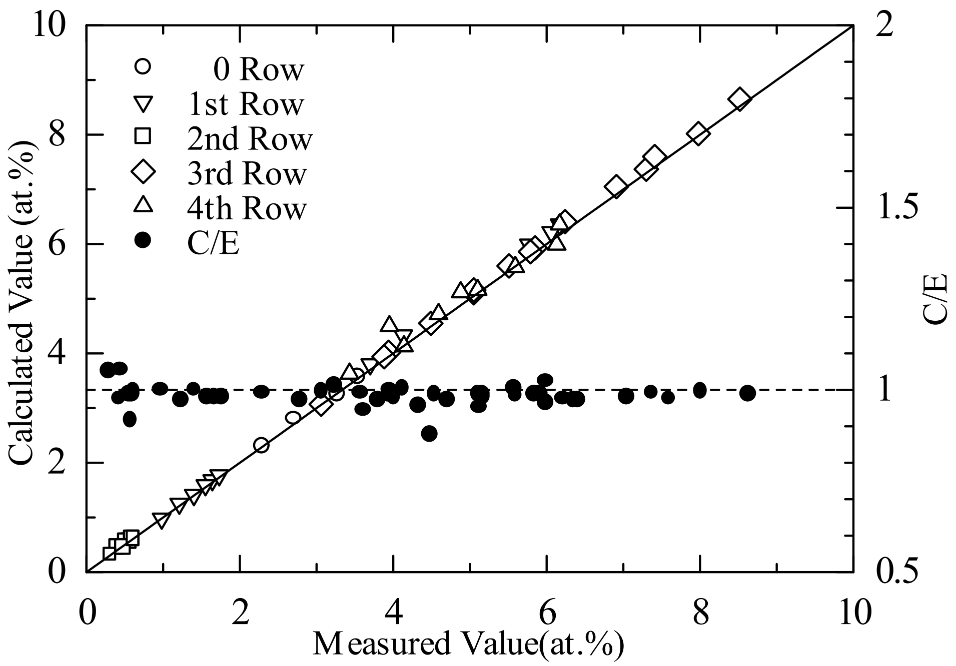


FIG. 8. Burnup ratio of JOYO MK-II driver fuel.

3.2.2.3. Decay heat measurement

The accuracy of decay heat calculations depends on the individual heat generation rate from fission product decay nuclides and actinides, and the burnup calculation for its production and transmutation. To obtain experimental data and to improve the accuracy of related calculations, the decay heat of MK-II spent fuel subassemblies was measured at the JOYO spent fuel storage pond [7]. The fuel burnup was approximately 66 GWd/t and the cooling time was between 40 and 385 days. The measured decay heat is shown in Fig. 9.

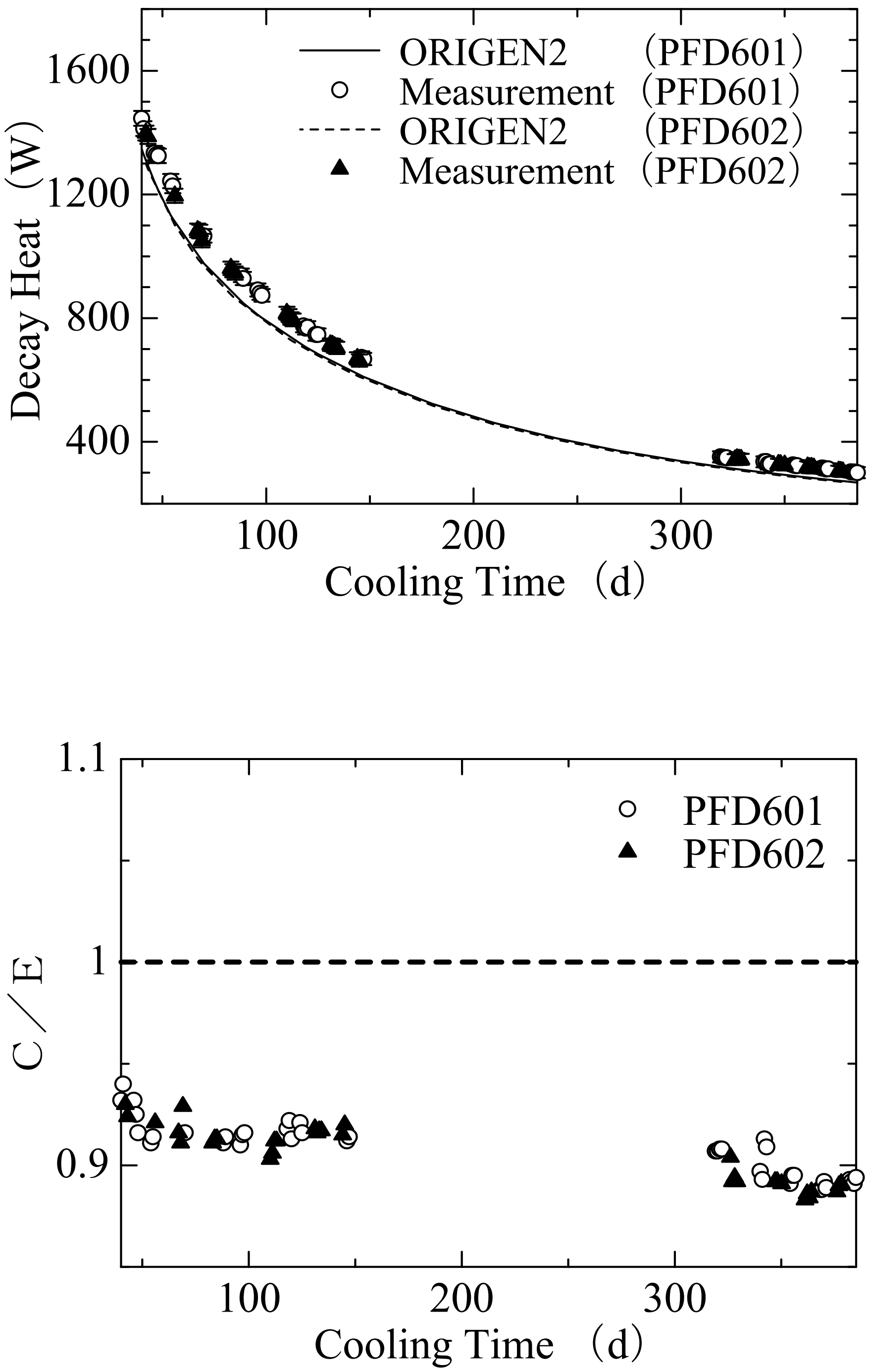


FIG. 9. Measured and calculated decay heat of JOYO MK-□ spent fuel.

The decay heat was then calculated with the ORIGEN2 code using the JENDL-3.2 cross section library and the JNDC-V2 [8] decay data and fission yield data library. The fuel power used as an input to ORIGEN2 was calculated by MAGI. The ratios of calculated to experimental values were between 0.94 and 0.89, and decreased with a longer cooling time as shown in Fig. 9. This systematic discrepancy needs to be further investigated, but the change with cooling time appears to be due to the actinides' decay heat uncertainty. Table 5 shows calculated and adjusted neutron flux and fluence.

TABLE 5. CALCULATED AND ADJUSTED NEUTRON FLUX AND FLUENCE

Item	Adjusted Value (NEUPAC)	MAGI	MAGI/ NEUPAC
(Core Region)	(1σ Error)		
ϕ_{total} (n/cm^2/s)	3.97×10^{15} (4.8%)	4.18×10^{15}	1.05
$\phi_{>0.1MeV}$ (n/cm^2/s)	2.61×10^{15} (8.0%)	2.83×10^{15}	1.08
DPA (dpa/s)	1.27×10^{-6} (5.0%)	---	---

3.2.3. Reactor dosimetry

The neutron flux calculation error rate was evaluated to be less than 5% in the fuel region according to the comparison between MAGI and reactor dosimetry test results (see Table 5). Figure 10 shows an example of adjusted neutron spectrum based on the foil activation method at the core center position of the MK-II.

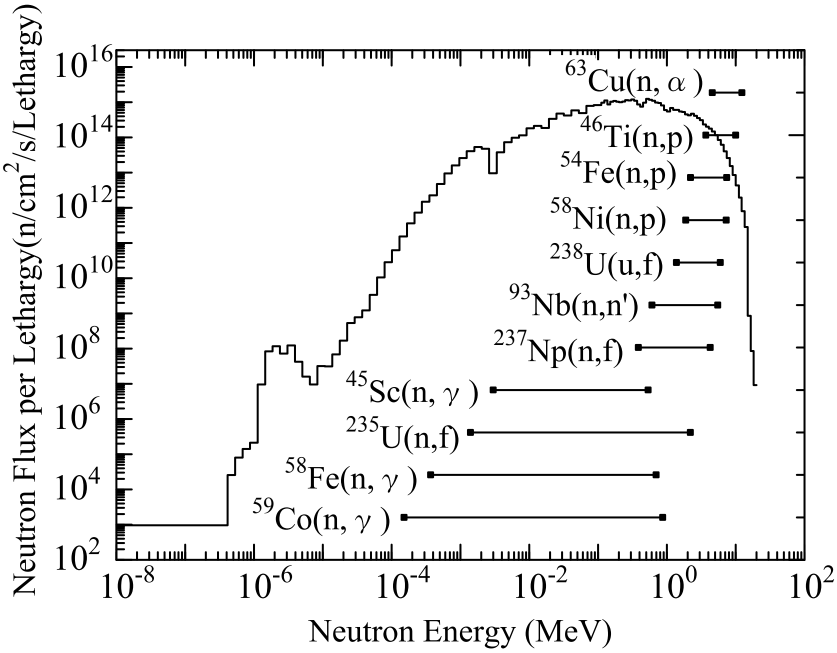


FIG. 10. Adjusted neutron spectrum at JOYO MK-II core center.

The dosimetry test results were used to correct MAGI calculations to assure the neutron fluence used in the post irradiation test analysis (see Table 5). A Helium Accumulation Fluence Monitor (HAFM) has been developed to improve the accuracy of long-term irradiation tests.

3.3. MK-II core characteristics database

During the MK-II operation, extensive data were accumulated from start-up and core characteristics tests. These core management and core characteristics data were compiled into a database [2]. The core management data includes core specifications and configurations, atomic number densities before and after irradiation, neutron and gamma flux, neutron fluence, fuel burnup, and temperature and power distributions. The core characteristics data include excess reactivities, control rod worths, and reactivity coefficients, e.g., temperature, power and burnup. These core characteristics data were recorded on CD-ROM for user convenience.

4. CHEMICAL ANALYSIS OF SODIUM AND COVER GAS

It is essential for steady and safe operation of a sodium cooled fast reactor to limit the coolant and cover gas impurities to prevent corrosion of reactor component materials and to reduce radiation dose by corrosion products (CPs). Therefore, impurity concentrations of both coolant sodium and cover gas argon were measured during the duty cycle operation and annual inspection period. The sodium impurity data include oxygen, hydrogen, nitrogen, chloride, tritium, metal elements and radioactive ^{110m}Ag, ^{22}Na, ^{137}Cs. The cover gas impurity data include O_2, N_2, CO, CO_2, H_2, CH_4, He, ^{3}H and radioactive xenon and krypton isotopes.

These data were measured by chemical analysis, gas chromatography, beta-ray scintillation and gamma-ray spectrometry. The sodium impurity concentrations were also determined by the sodium temperature in the plugging indicator. As an example, the trend of oxygen and hydrogen in the primary sodium are shown in Fig. 11.

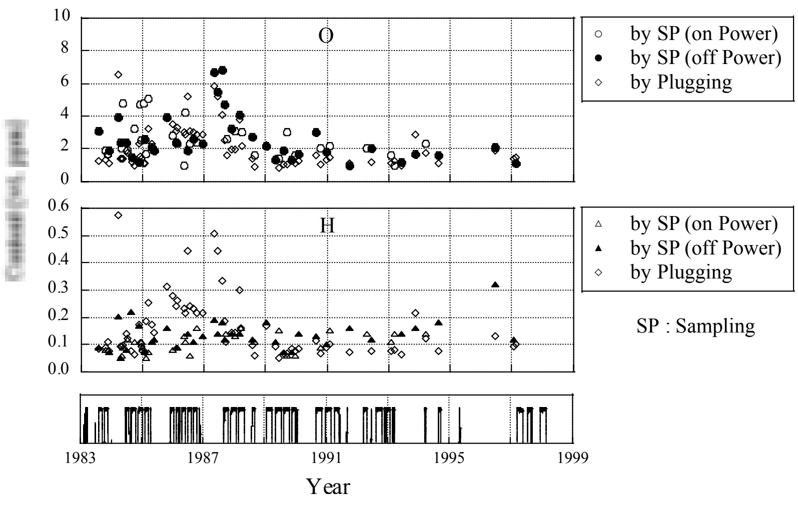


FIG. 11. Hydrogen and oxygen content in JOYO primary coolant sodium.

5. DECAY HEAT REMOVAL TEST OF NATURAL CONVECTION

The inherent safety of a fast reactor can be demonstrated by proving the natural convection capability and establishing analytical techniques based on experiments. A series of natural convection tests was performed from 1981 to 1986 in JOYO. The natural convection test from 100 MWt, which is the most critical situation for the reactor, was carried out at the end of 12^{th} cycle of the MK-II core in 1986 [9]. The test was initiated by tripping primary and secondary sodium pumps manually without pony motor operation, and the reactor was shut down simultaneously by tripping the pumps; moreover, the blowers of the DHXs were stopped immediately with the reactor scram signal. The central driver fuel subassemblies undergo the most severe temperature transition during the tests. Figure 12 shows the outlet temperature of one of these assemblies, together with primary coolant flow rate variations. The peak temperature reached 519°C due to coolant flow rate reductions. This peak is significantly below the initial temperature of 548°C. Consequently, it is shown that the temperature increase will not cause any safety-related problems, such as fuel cladding failure.

The post analysis results from a plant wide dynamics code MIMIR-N2 are in excellent agreement with the experimental data as shown in Fig. 12. Various key parameters are clarified to improve the calculation accuracy through the study. In the short-term analysis, the evaluation of thermo-hydraulic behavior in the core is largely affected by the inter-assembly heat transfer effect, the pump flow coast characteristics and coolant flow distribution. For the long-term analysis, it is important to assess precisely the buoyant head effect in the IHX, the heat exchange effects in the lower plenum of the IHX and others. The experimental result is also applied to the assessment of natural convection characteristics in the MONJU reactor.

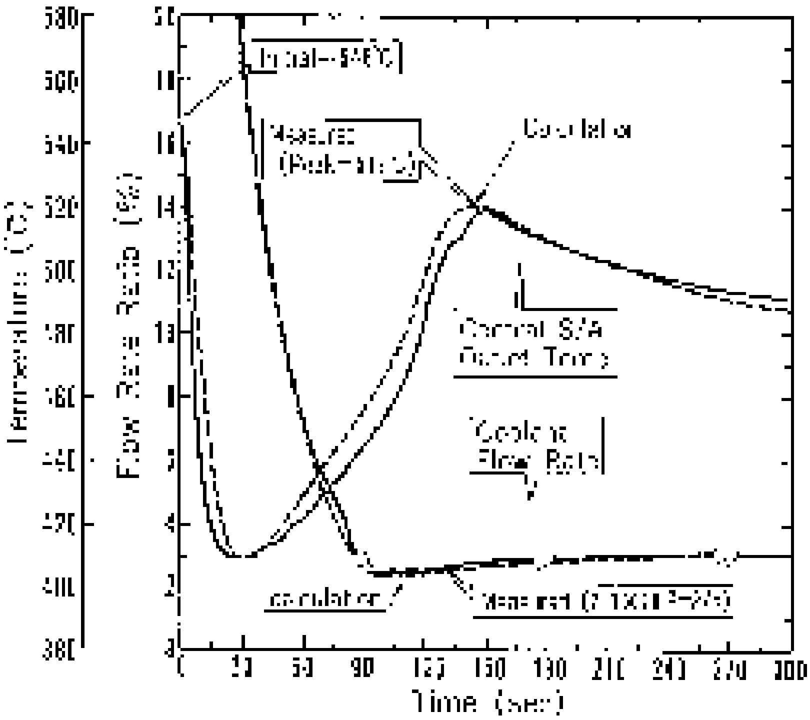


FIG. 12. Test results and analysis of neutral convection test.

6. UPGRADE OF FFD SYSTEM AND INSTALLATION FP TRAPS

The fuel failure detection (FFD) and the failed fuel detection and location (FFDL) are important for LMFBR plants to achieve high availability and operational reliability. Fission product (FP) traps are important for safety reasons: especially for conducting safe Run-beyond-cladding-breach (RBCB) tests.

The JOYO FFD system consists of both delayed neutron (DN) monitoring systems and a cover gas (CG) precipitating system. The schematic diagram of the JOYO FFD system is illustrated in Fig. 13.

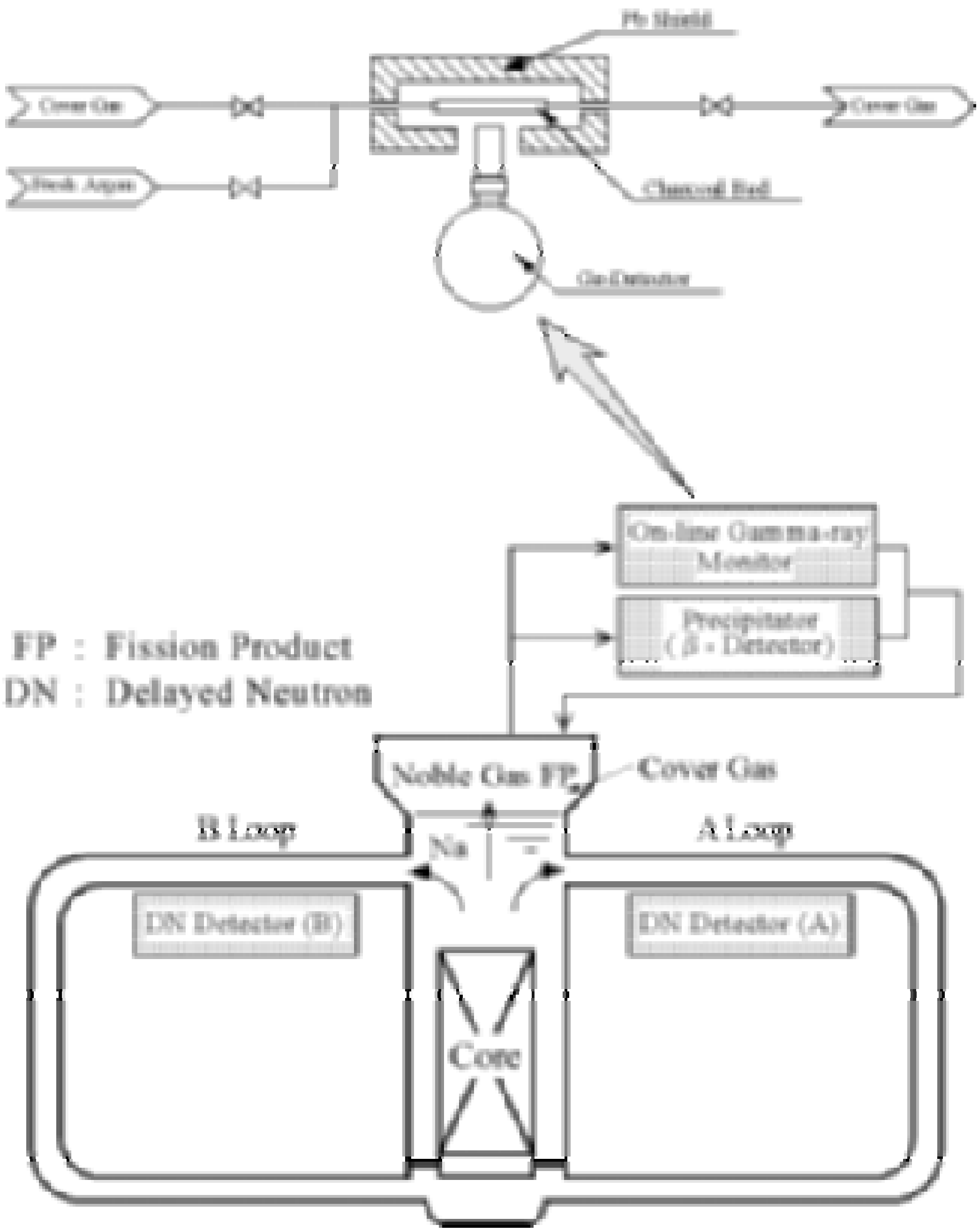


FIG. 13. Schematic diagram of JOYO FFD system and OLGM.

Ttwo DN monitoring systems are located adjacent to the primary cooling loops to detect the delayed neutrons emitted from precursors released into the coolant sodium. The CG precipitating system detects fission product of ^{88}Rb i.e. beta decay of ^{88}Kr released into the cover gas argon.

A Run-to-Cladding-Breach (RTCB) test is planned in JOYO. The RTCB test is expected to improve the FBR fuel performance. The results will increase the fuel burnup and extend the cladding life-time. As part of the preparation work, the FFD system has been upgraded to improve its accuracy and reliability and FP traps have been installed. A series of simulated fuel failure tests has been conducted [10].

6.1. Upgrade of the FFD system

6.1.1. On-line gamma-ray monitor

The On-line Gamma-ray Monitor (OLGM), shown in Fig. 13, has been developed and installed in JOYO.

The OLGM consists of a charcoal bed that is made to selectively adsorb the isotopes of krypton and xenon, and a high purity Ge detector. The special feature of this system is to be able to identify the isotopes in the cover gas by means of gamma-ray spectrum analysis. The OLGM is also used to detect the tag gas that is originally contained inside a pressurized stainless steel capsule. In the irradiation environment, the creep rupture may cause the capsule to fail and release the tag gas into the cover gas. As shown in Fig. 14, the activated tag gas nuclides among the background fission products are clearly detected by OLGM and this method was found to be applicable [11].

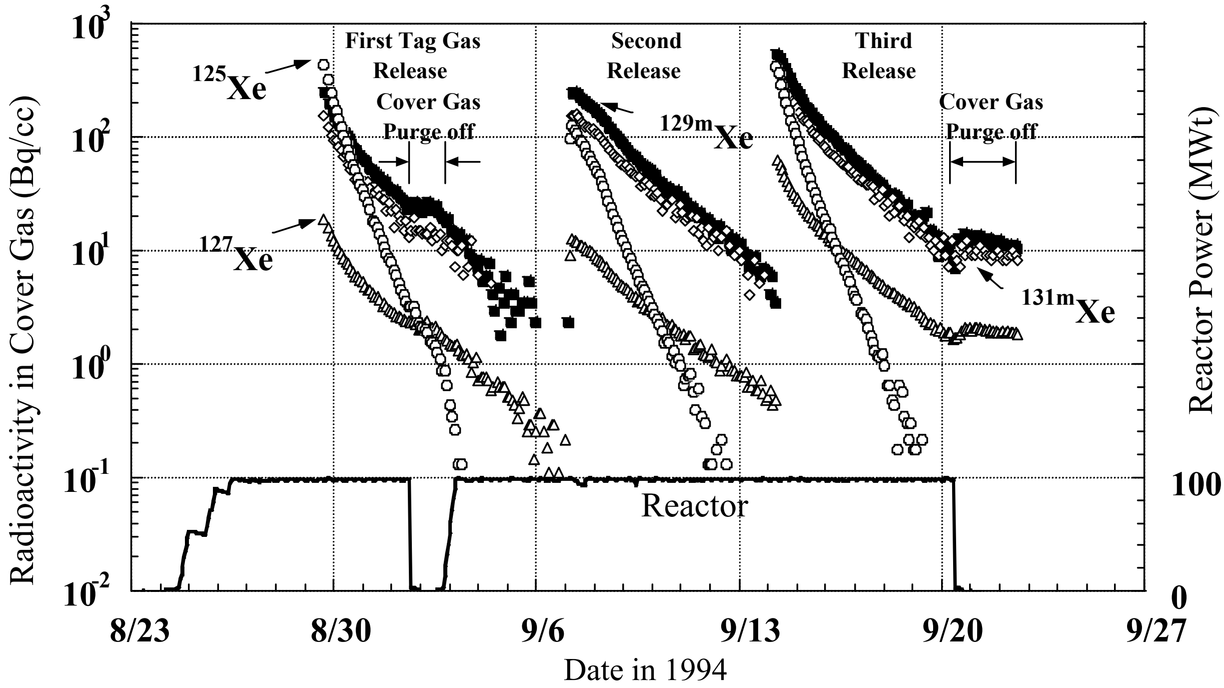
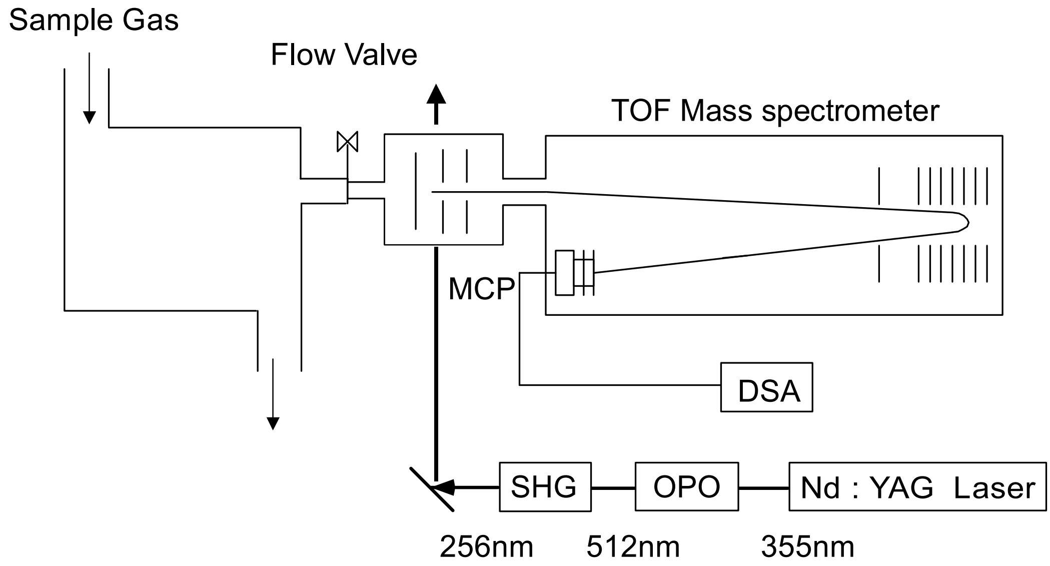


FIG. 14. Measured tag gas activation products in JOYO.

6.1.2. Resonance ionization mass spectrometry

The Resonance Ionization Mass Spectrometry (RIMS) is a method to detect xenon and krypton with ultra high sensitivity using a laser technique [12] developed in collaboration with the University of Tokyo and Nagoya University. The block diagram of a RIMS system with a time of flight (TOF) mass spectrometer is illustrated in Fig. 15.



RIMS: Resonance Ionization Mass Spectrometry
OPO : Optical Parametric Oscillator
SHG : Second Harmonic Generator
MCP : Micro Channel Plate
DSA : Digital Signal Averager

FIG. 15. Block diagram of RIMS system with TOF mass spectrometer.

The optical parametric oscillator (OPO) is excited using the YAG laser at the 355 nm wavelength. The higher harmonic wave is then generated from 512 nm OPO light in the secondary higher harmonic generator (SHG), and the laser-ionized sample gas flows into the ionization chamber. The ion is then detected in the micro channel plate (MCP), and it is counted by the digital signal averager (DSA). An example of measured data for sample argon gas containing 10 ppb of xenon with a natural isotopic distribution is shown in Fig. 16. A sharp mass spectrum was obtained by RIMS, which demonstrates a high sensitivity to the diluted cover gas in the order of few ppb.

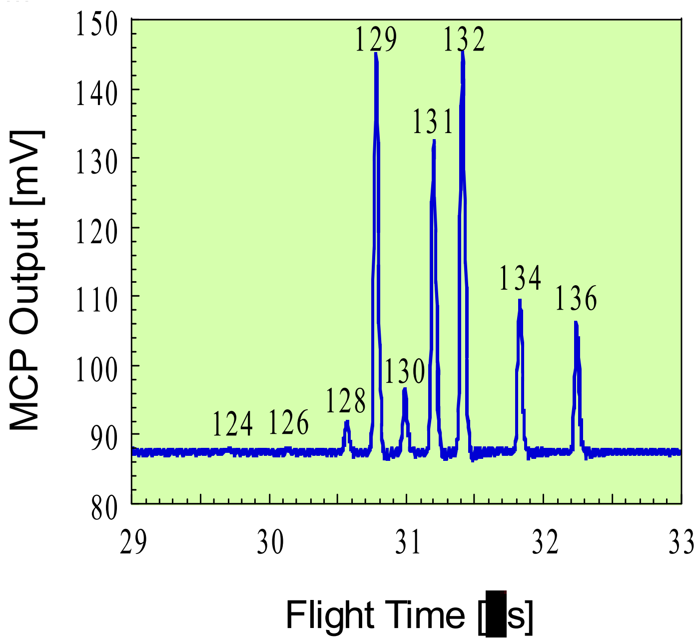


FIG. 16. Measured mass spectrum of natural xenon.

6.2. Installation of FP traps

Two types of FP traps have been installed in JOYO. One is a cesium trap installed in the primary coolant sodium purification system to capture cesium released from failed fuels. An open pore, foam-like glassy carbon that consists of thin struts of Reticulated Vitreous Carbon (RVC) is used as a material for collecting cesium. The capacity of this trap is designed to be 7.4E+12 Bq. The other trap is a Cover Gas Clean-up System (CGCS) to collect and store the noble fission gas released from failed fuels. Although it is planned that only one failed fuel pin will be in the core at any time, the CGCS is designed to handle the releases of up to twelve failed fuel pins.

6.3. Fuel failure simulation tests

An in-pile simulation test was carried out at the end of the 7th operational cycle using two artificially defected (slit) fuel pins to verify the performance of JOYO's FFDL system. This system uses the sodium sipping method. The slit is 1.0 mm long and 0.1 mm wide and perforated on the fuel cladding at gas plenum position. In the FFDL operation, a signal level of the test subassembly was several hundreds times higher than the background measured for other core subassemblies (Fig. 17). With this test, the FFDL system was confirmed to have the capability to identify the failed fuel with a defect at a gas plenum position. A FP source using U-Ni alloy tubes was irradiated at the end of 15th operational cycle. The test results show two important results. Both the CG and DN monitoring system were successfully calibrated and the CG monitor was confirmed to have enough sensitivity to the FP gas released from the failed fuel pins. The major constants for the cover gas and DN behavior models were determined and the disengagement rate constant of the FP gas from the sodium coolant to cover gas region varied depending on the flow rate of the primary cooling system.

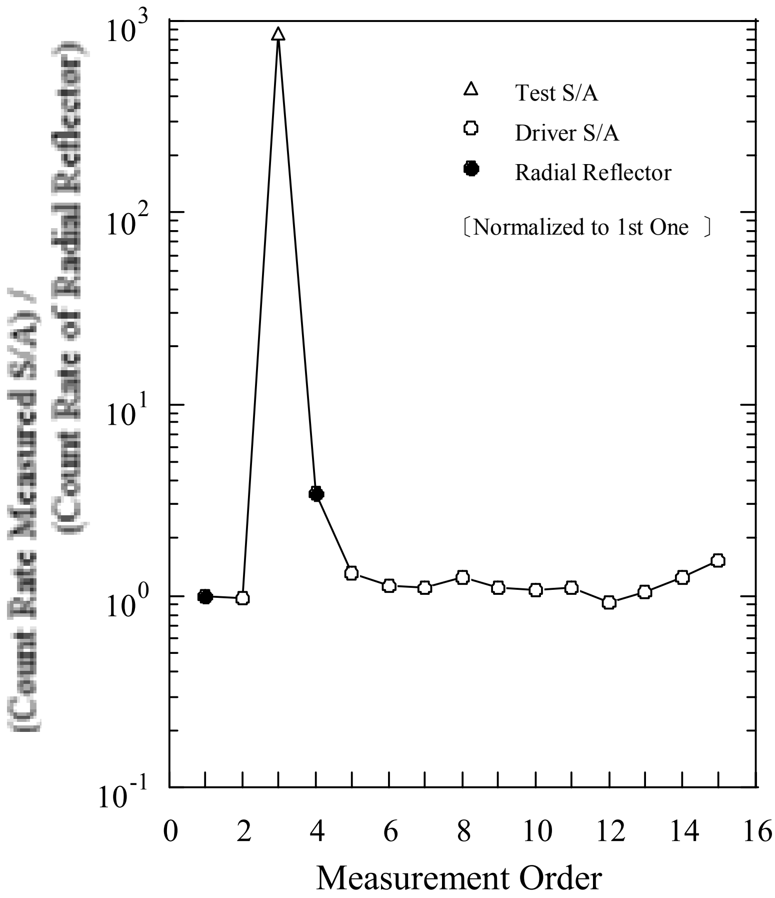


FIG. 17. Relative count rate of Xe-133 gamma-ray from examined subassembly.

7. MEASUREMENT OF RADIOACTIVE CORROSION PRODUCTS (CP)

The sodium cooled fast reactor JOYO has been operated more than 20 years (about 5 years of effective full power years) since its initial criticality and the cumulative reactor output achieved over 1.9E+5 MWd. Since JOYO has not yet experienced any operation with breached fuels, FP radioactive contamination has not become an issue in the plant system. To reduce the radiation dose from long-lived ^{22}Na, all primary coolant sodium in the main circulating loops is drained into a storage tank during annual plant inspections. Under these conditions, the spatial gamma-ray dose rate distribution is dominated by the radioactive CPs deposited on inner surfaces of the primary piping and components. This means that most personnel dose was due to these CPs.

7.1. Measured results and analysis

The CP deposits on the inner surfaces of the primary main piping have been measured at every annual inspection period since the end of MK-I operation in July 1982. These measurements are made at 14 locations, shown in Fig. 18, using a Ge solid-state detector system. The detector system was calibrated with a piping mock-up using two planer type standard gamma-ray sources, ^{54}Mn and ^{60}Co, so that the absolute amounts of CP deposits could be obtained from the measured gamma-ray spectra.

In every annual inspection, gamma-ray dose rates from these CP deposits have been measured using calcium sulfate (CaSO4) thermo-luminescence dosimeters (TLDs). The gamma-ray dose rate distribution near the piping is measured in detail at 93 locations at one-meter intervals along loop (A) from the outlet to the inlet of the reactor vessel. At each location, four TLDs are placed every 90 degrees around the thermal insulator cover. The geometrical conditions for the measurements are almost the same as those for the radioactive CP deposit mentioned above.

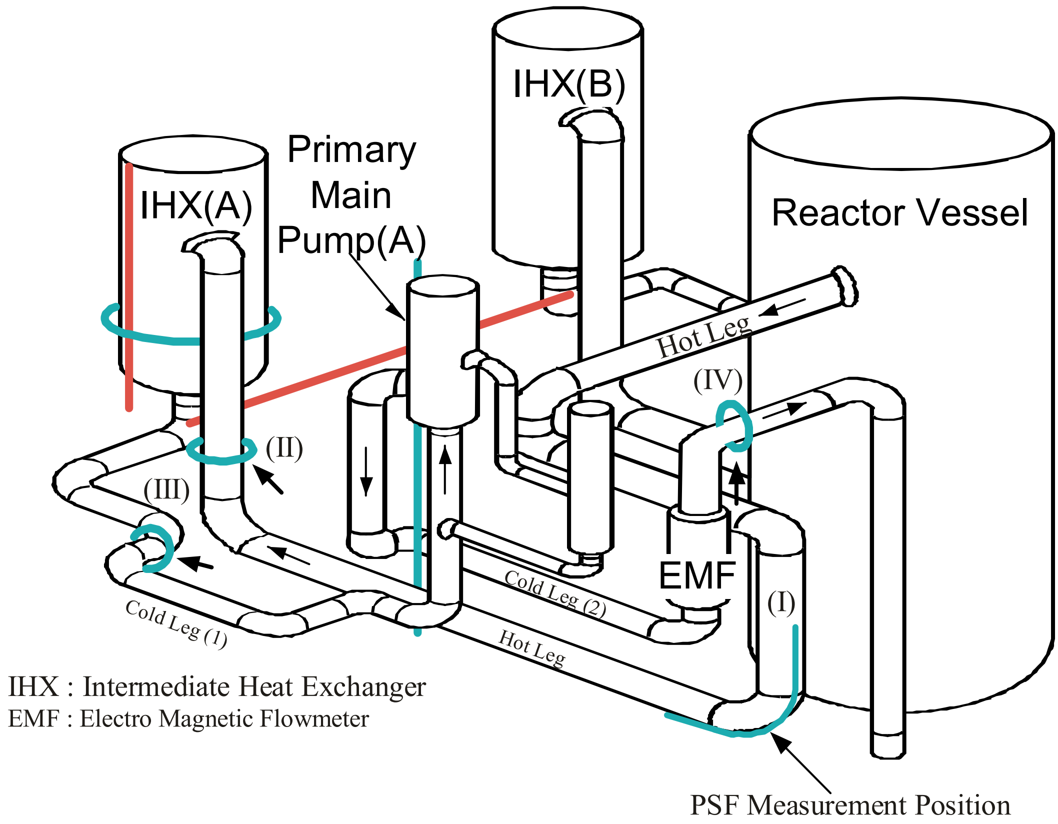


FIG. 18. Primary cooling loop of JOYO.

The CP behavior analysis code PSYCHE [13] has been developed and verified with the measured radiation data to analyze the distribution of corrosion product in the primary cooling system. A radiation dose calculation code has been developed by JNC, to analyze the CP deposition distribution along the piping and components of the primary cooling system. The build-up of CPs is shown in Fig. 19 together with the reactor operation time. The PSYCHE calculations agree with the measured values.

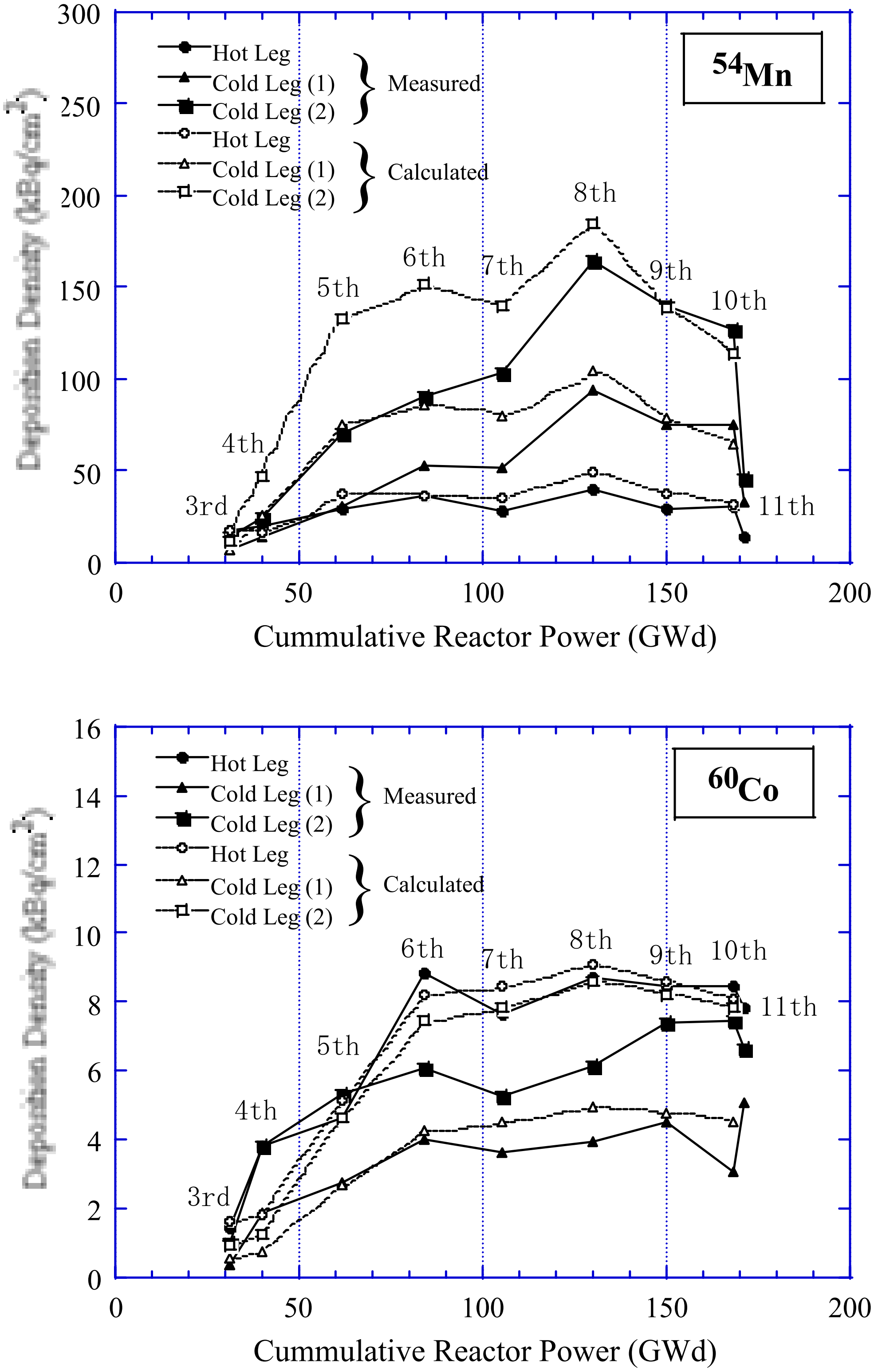


FIG. 19. Comparison of measured and calculated CP build-up in primary cooling piping (A).

7.2. Application of optical fiber

A Plastic Scintillation Fiber (PSF) measured the dose rate distribution in the primary cooling system of JOYO [14]. Figure 20 shows the schematic diagram of the PSF system.

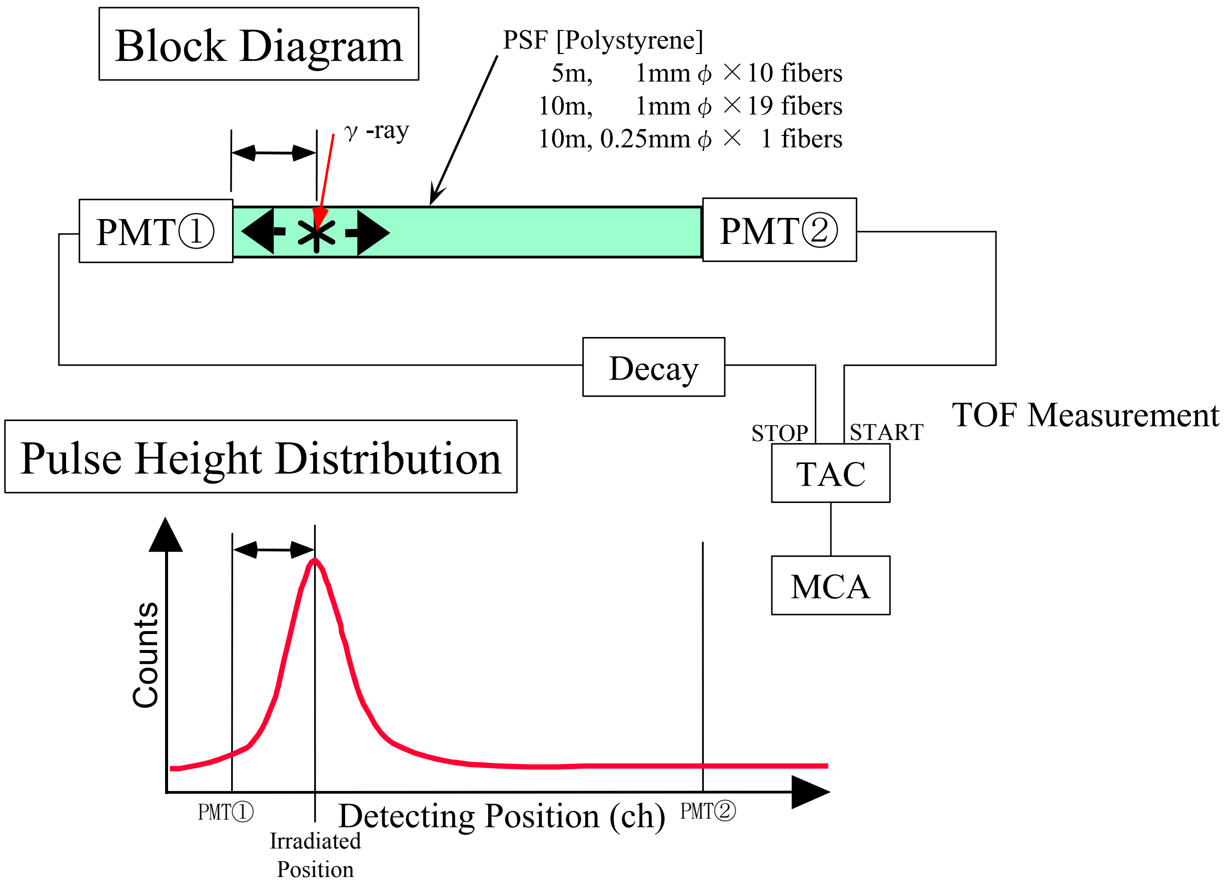


FIG. 20. Block diagram of radiation measurement using PSF.

Polystyrene was used as a scintillator that emits light in response to gamma-rays. The radiation source's location was determined with the time of flight method by measuring the time interval between the signals from two photo multipliers at each end of PSF. As seen from the pulse height distribution in Fig. 19, the measured count rate distribution does not have a sharp gradient even for a single point irradiation; therefore, the unfolding method is applied to reproduce the actual dose rate distribution.

The gamma-ray dose rate profile of the A loop IHX, shown in Fig. 21, is an example of the measured PSF data. Two peaks were observed due to horizontal plates in these positions where there were large CP deposits. Comparing PSF results with a series of TLD point data, large differences were observed at these peaks. However, by employing the unfolding method, the PSF data coincided with the TLD results. The gamma-ray dose rate distribution measurement was greatly improved by the use of PSF.

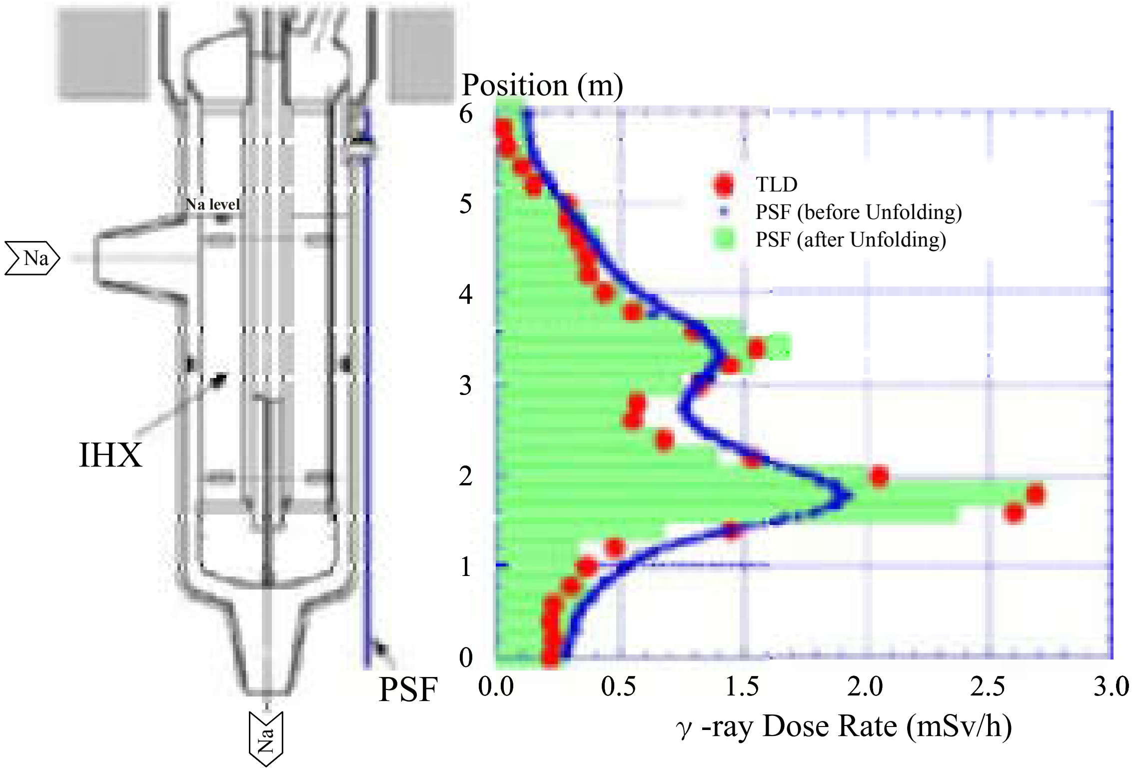


FIG. 21. Measured axial gamma-ray dose rate distribution of IHX.

8. OPERATION AND MAINTENANCE SUPPORT SYSTEMS

The JOYO operation and maintenance support systems ensure more stable operations and improve operational reliability. Artificial intelligence techniques [15] have been applied to develop these systems. One system objective is to support intelligent decision making by the operators and maintenance engineers, and another is to conduct skill-based and rule-based operator actions automatically. With proper instructions and guidance from the support system, the JOYO operators can make better decisions and carry out necessary actions with more confidence and less mental pressure.

8.1. Operation support system

The operation support system named JOYCAT (JOYO Conducting and Analyzing Tool) [16] was developed to help operators make intelligent and quick decisions in cases of anomalous event occurrences. Figure 22 illustrates the JOYCAT hardware configuration. The JOYCAT system consists of a knowledgebase and an inference system connected to the JOYO data acquisition system (JOYDAS). The JOYDAS collects on-line plant operation data from several thousands sensors located in different positions in the plant at interval of 0.25 second. Prior to its application to the JOYO plant, JOYCAT was validated by a full scope operator-training simulator. The alarm signals used for this validation were triggered by a manual scram during the reactor shut down process and by activating the reactor safety system during a periodic test. The system has been used since 1988.

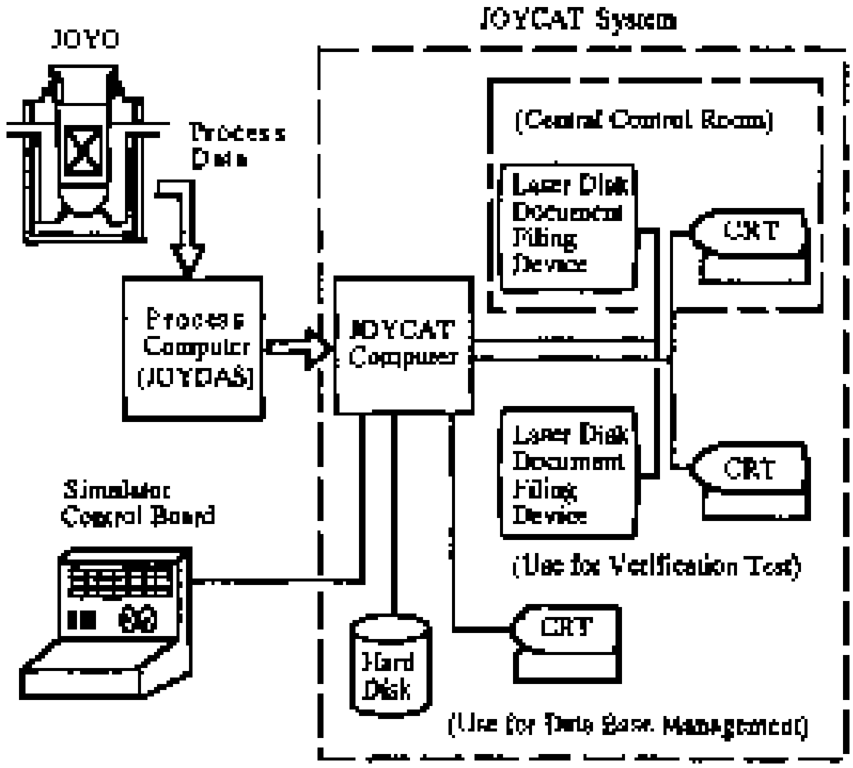


FIG. 22. JOYCAT hardware components.

The features of the JOYCAT system are:

- When an alarm occurs, the system diagnoses the plant conditions based on data from the reactor systems and the main components.
- Based on the system knowledge base, the alarm sequences and plant interlocks are checked. If nothing wrong is found, the system identifies the cause of the alarm. When an anomaly is found in some sequence or a safety system failure, the related information is displayed quickly on a CRT for the operators.
- After the cause of the alarm is identified, the most suitable operation manual for the anomaly is displayed for the operators.

8.2. Automatic control rod operation system

The JOYO operators control the reactor power, i.e. neutron flux level, by adjusting the position of the control rod subassemblies in the core. This is a manual operation performed from the central control room. To improve operational reliability as well as to reduce the mental load on the operators, an automatic control rod operation system [16] has been developed. This system has the following capabilities:

1) Drive the control rods;
2) Indicate the rod drive stroke for each control;
3) Indicate the criticality point (5E+4 cps for Source Range Monitor);

1) Drive the control rods;
2) Indicate the rod drive stroke for each control;
3) Indicate the criticality point (5E+4 cps for Source Range Monitor);
4) Monitoring of the reactor period, neutron flux, thermal output of the reactor, rate of reactor power change, heat-up rate at the reactor vessel inlet, and the temperature difference between the overflow tank and the reactor vessel outlet;
5) Guide the plant operation;
6) Display information concerning the rod operation and the current plant conditions, i.e. trend graphs of the thermal power, neutron flux and reactivity of the core.

In actual operation, the operator's actions in accordance with the above guidance should be conducted in a different manner depending on the reactor power level as described in Figure 23. A fuzzy algorithm based on linguistic rules is employed to control non-linear characteristics, whereas this is a difficult problem for the conventional PI controller.

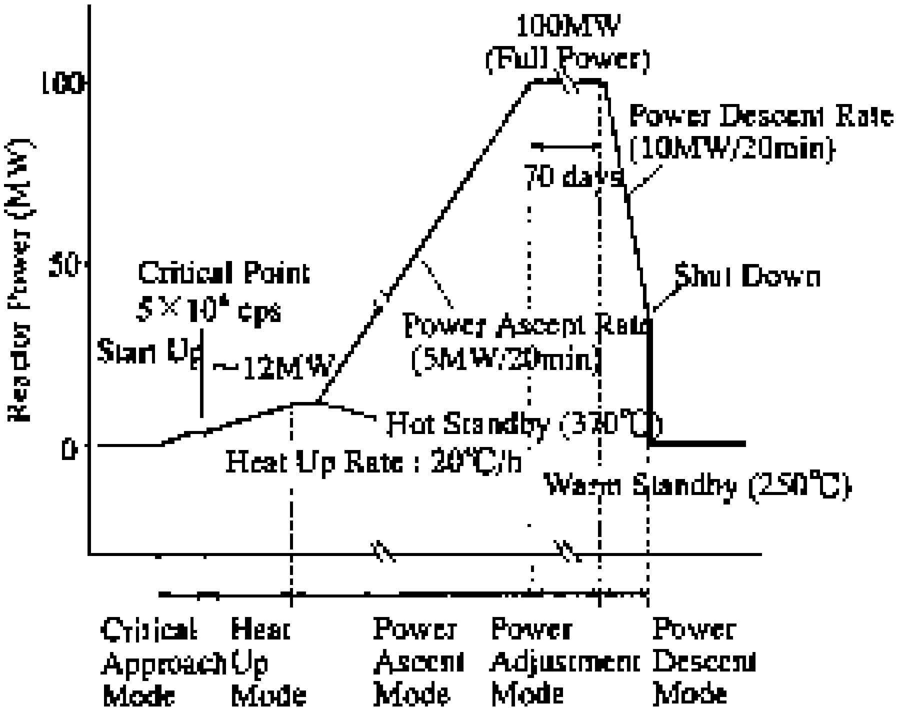


FIG. 23. Control mode of automatic control rod operation system.

In the past, the control rod drive stroke was calculated by the operators prior to manual control when approaching criticality. After the installation of the system, it calculates and displays an inverse multiplication curve and the stroke of each control rod subassembly that needs to be driven using the neutron flux and the vertical control rod position. The reliability of this system was validated during a complete operational cycle of the MK-II core. It was also demonstrated that the operation guides provided by the system were very similar to those chosen by the experienced JOYO operators.

8.3. Maintenance support system

High security and reliability of equipment is required in nuclear reactor plants. The equipment is designed based on failure analysis and preventive maintenance and condition monitoring are recommended. The condition monitoring technique has been developed to detect the failure and degradation of the machine as soon as possible to diagnose anomaly causes and to evaluate the machine's damage. The equipment is used under various stresses depending on their operating environments and these stresses cause degradation or failures such as fatigue, wear and tear, corrosion and others. In most cases, these appear as changes in vibration or acoustic noise so vibration monitoring is a popular and effective technique especially for rotating machines like motors or pumps.

In JOYO, a vibration monitoring systems named MEDUSA (MEchanical-fault Diagnosis Using Spectrum Analysis) [17] has been developed as an on-line vibration monitoring system for the major rotating machines such as the main pumps of the primary and secondary cooling systems. MEDUSA monitors the vibration of major rotating machines automatically and continuously, and it notifies plant operators and maintenance engineers of any anomalies it detects. Furthermore, the MEDUSA could assist in the analysis and interpretation of the vibration data.

8.3.1. System configuration

Figure 24 illustrates the system configuration of MEDUSA.

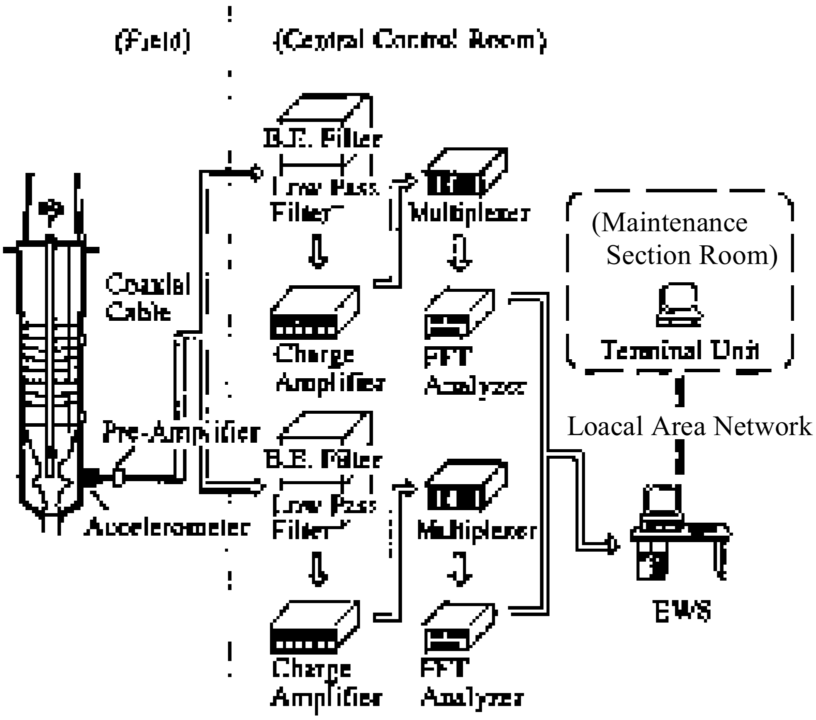


FIG. 24. System configuration of MEDUSA.

Vibration signals are collected by accelerometers mounted on each rotating machine and the signals are sent to the multiplexers (Fast Fourier Transform (FFT) analyzers) through the amplifiers and filters. The engineering work station (EWS) manages the instruments and vibration data. The FFT analyzers and multiplexers save the CPU load of the EWS for the signal processing so MEDUSA could easily deal with the increasing number of monitoring machines.

The MEDUSA monitors vibration levels (r.m.s. and peak levels), auto power spectrum densities (APSDs), waveforms and probability density functions at every hour automatically, then the r.m.s. levels and the APSDs are compared with acceptable vibration levels to judge whether the machine's condition is normal.

If an anomaly condition is detected, warning messages and beeps notify the plant operators and maintenance engineers. Monitoring intervals are variable and it is possible to monitor more frequently when an anomaly symptom is detected or for test runs after an inspection.

Vibration data has been kept for 300 days on the EWS, and the previous data is dumped into digital audiotapes. These data could be accessed on the terminals anytime, so that the follow-up survey over a long term is possible. Operators can review how an anomaly symptom appears and how it increases by a trend graph and a three dimensional graph (time-frequency-amplitude of vibration).

The characteristic vibration data for an anomaly condition of the rotating machine is saved with a label and description that includes phenomena, causes and measures against the anomaly. These data could be used as a reference to judge the machine condition and to make a necessary action.

8.3.2. Experience of vibration monitoring in JOYO

The MEDUSA has been used as a regular part of plant maintenance and operation since 1990, and it is useful to detect anomalies and to diagnose their causes. As an example of vibration monitoring in JOYO, the vibration level of a cooling blower suddenly increased by a factor of ten on 31st December 1992. This blower is placed in a nitrogen gas atmosphere and cannot be repaired while the reactor is operating.

Figure 25 shows the APSD of vibrations before and after the increase of vibration. The vibration increased in higher frequency region over 5 kHz (increased -50 dB to -20 dB around 6 kHz). Therefore, it was estimated that the vibration occurred due to loss of lubrication oil or bearing failure.

Later at the annual inspection of the blower, it was found that there was a loss of lubrication oil in the bearing box and the vibration level decreased after refilling the oil.

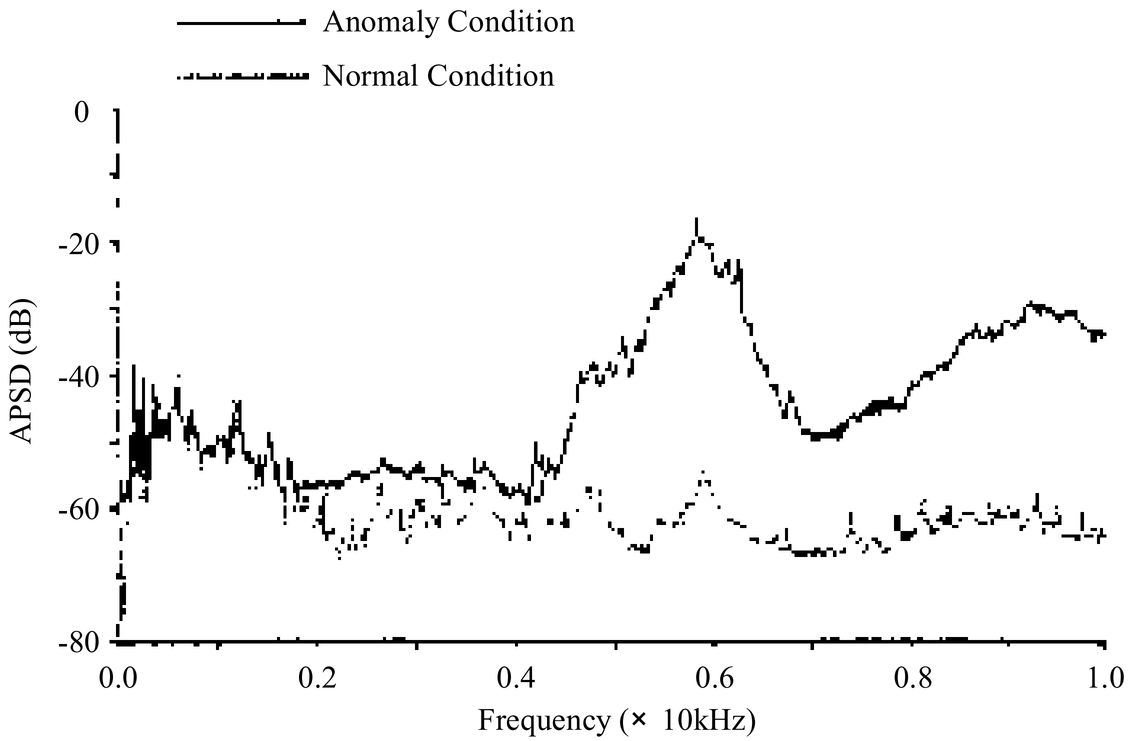


FIG. 25. Comparison of vibration spectra (loss of lubricating oil).

9. IMPROVEMENT OF COUNTERMEASURES AGAINST SODIUM LEAKS

A sodium leak accident occurred in the MONJU secondary cooling loop in December 1995. At that time, the 11th annual inspection was being performed at JOYO and an inspection of sodium piping and components was immediately carried out to confirm their integrity and to verify that there were no sodium leaks. After the MONJU accident, an investigation was completed by the Safety Authority and findings on the cause of the sodium leaks and ways to mitigate their effects were published.

According to this report, the structural integrity of the JOYO thermocouple well was tested and confirmed by hydraulic vibration evaluation based on water flow tests and ASME standards. Modifications were then made at JOYO to improve countermeasures against sodium leaks in the secondary loop [18].

The improvements of these sodium leak countermeasures in the secondary loop stressed prevention, early detection and mitigation of the effects. The improvements were determined according to the following considerations:

(1) Sodium leak prevention

- All the thermocouples in the secondary loop were inspected by non-destructive examination (X-ray, fiber scope inspection) to confirm their integrity.
- All the thermocouples were replaced by new ones with compression fittings that seal any sodium leaking between the thermocouple well and sheath in the event of a thermocouple well rupture.

(2) Early detection of sodium leaks

- A new sodium leak monitor was installed in the central control room to verify the function of each sodium leak detector.
- A new fire alarm monitor was installed in the central control room to verify the function of each fire detector. This monitor has a second alarm function that re-alarms if another detector goes off.
- Monitor cameras were installed in the secondary cooling facility to view the sodium piping and components from the central control room.

(3) Mitigation of sodium leak effect

- Sodium leak trays were added under the sodium piping to prevent dispersion of leaking sodium.
- Anti-smoke dampers were added in the ventilation system to prevent the diffusion of sodium aerosol. The dampers in the ventilation system are interlocked with the sodium leak detector and fire detector to shut automatically.
- The penetration gaps between the piping and walls were sealed to prevent the diffusion of sodium aerosol.

(4) Improvement of Manuals

- Upgrade of Operational Manuals.
  i. The emergency operational manual for JOYO sodium leaks was reviewed and verified as valid.
  ii. The emergency operational manual was upgraded by incorporating the improved countermeasures against sodium leaks.
- Revision of education and training schedule
  ii. Emergency operation training, sodium fire-extinguishing training and overall disaster prevention training have been carried out.
  iii. The education and training schedule was revised to provide more skilled training.

JOYO's safety practice for sodium leaks were investigated by the Safety Authority. After a successful investigation, JOYO was allowed to restart operations in March 1997.

10. UPGRADING PROGRAM AND MODIFICATION OF CORE AND COOLING SYSTEM COMPONENTS

JOYO is expected to play a greater role in providing an irradiation field for fuels and materials irradiation tests and for the demonstration of innovative safety related systems for future fast reactors. To meet the increasing requirements for the various irradiation tests, the JOYO MK-III upgrading program was initiated to improve its irradiation capability [18 – 20]. The main objectives of the MK-III program are to increase the neutron flux, modify the

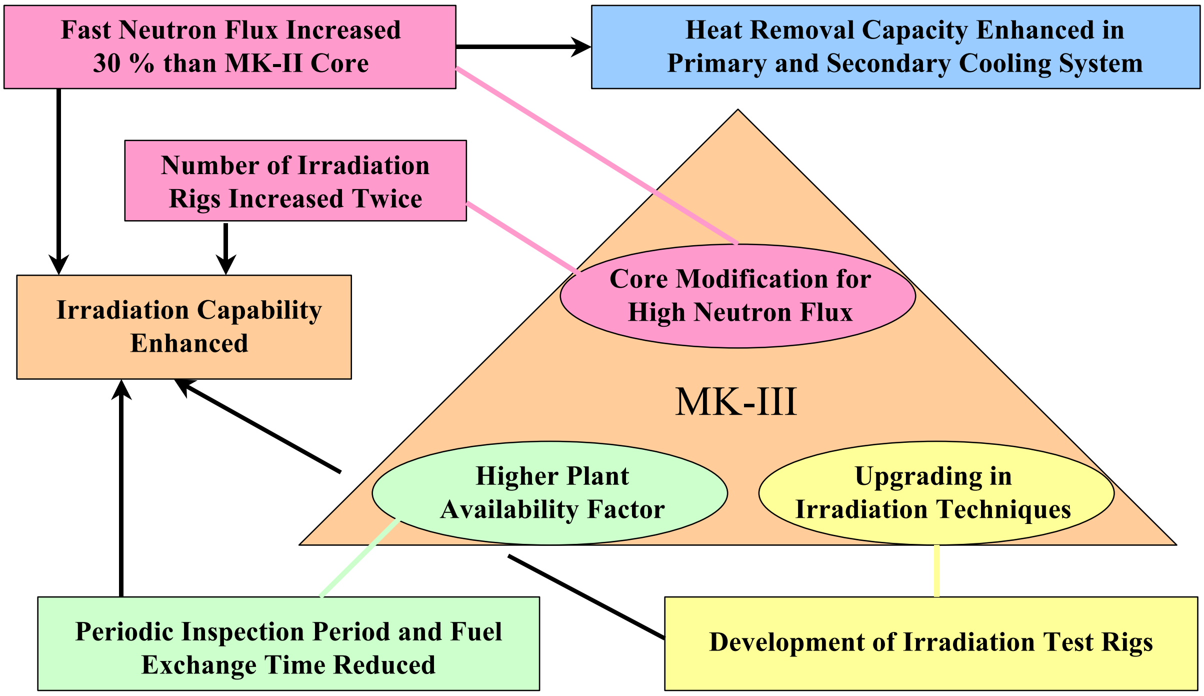


FIG. 26. Main purpose of MK-III project.

10.1. Outline of MK-III project and specification

The basic specifications of JOYO are previously shown in Table 1, which compares the MK-II and MK-III cores. The MK-III core is divided into two regions with different plutonium contents and the core height is decreased from 55 to 50 cm to obtain higher neutron flux with smaller power peaking. Two of six control rods are shifted to the edge of the outer core to enlarge the high neutron flux irradiation field. Two layers of radial stainless steel reflector are replaced by the shielding subassemblies, which contain boron carbide pellets, to reduce the neutron dose in the radial direction.

A whole plant design optimization increases the reactor thermal output from 100 to 140 MWt. The fast neutron flux increases about 30% as shown in Fig. 27 and the maximum allowable number of fuel irradiation test rigs increases from nine to twenty-one. The MK-III core will support various irradiation tests on advanced fuels like MA doped fuel, high plutonium content MOX fuel and vibration packed fuel. The irradiation technology has also been developed to expand the capability and flexibility of fuels and materials irradiated.

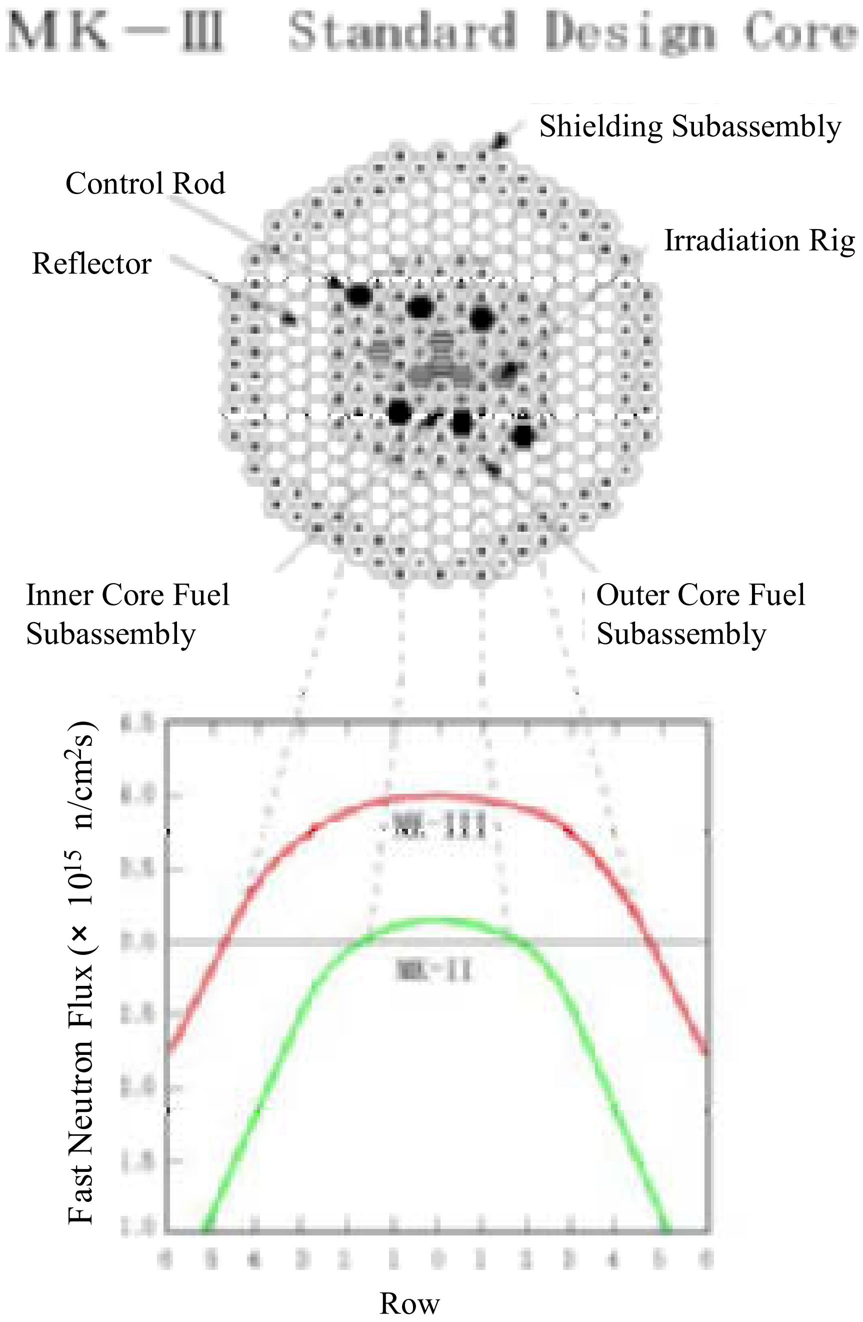


FIG. 27. Core configuration and fast neutron flux distribution in MK-III core.

The JOYO cooling systems need to be modified to increase the heat removal capacity. Figure 28 shows the components to be exchanged in the modifications and compares the plant condition in the cooling system. The primary system sodium coolant flow rate increases by 20% and all Intermediate Heat Exchangers (IHX) and Dump Heat Exchangers (DHX) were replaced. The cooling system modification completed in September 2001 and the cooling systems were refilled with sodium then the initial purification was conducted. The cold trap in the secondary sodium purification system will be replaced in the autumn of 2002.

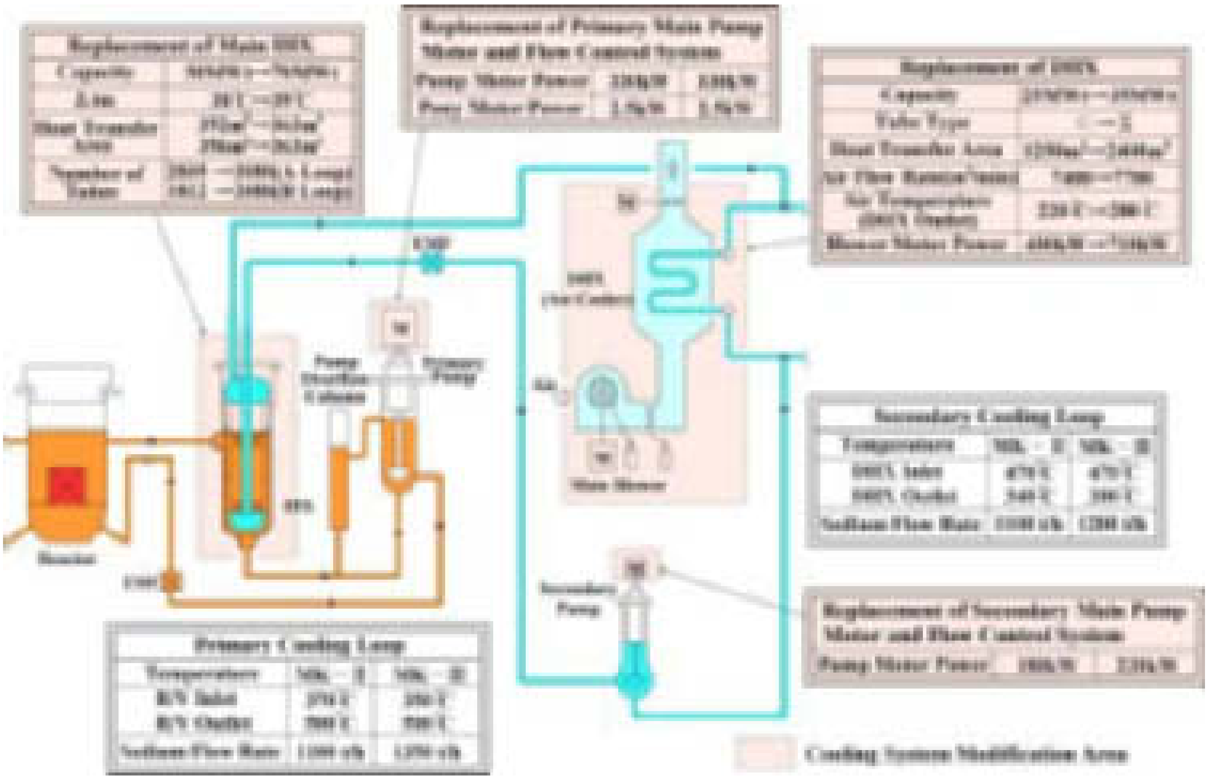


FIG. 28. Cooling system components to be exchanged.

10.2. Modification work experience

Special attention was given to prevent air containment when cutting the sodium boundary during this modification work. The following procedure was employed for replacing the DHX. First, the temporary supports for sodium piping and argon gas supply lines were installed in preparation for cutting the sodium piping. Next, the sodium piping was cut to two-thirds of their thickness using the cutting tools. A glove box was installed to cover the sodium piping cuts to keep the cover gas boundary filled with argon.

The sodium piping was cut off after a leak test was conducted. The sodium on the inner surface of the cutting part was scraped off and the residues were rinsed with alcohol. After the shut-off plug was set in the inner part of the piping to completely shield it, the beveling of the welding part was done to prepare for the new piping weld. Since all the cutting positions of the sodium piping are horizontal, the cutting powder from the piping is relatively easy to collect, preventing it from entering in the cooling system. In this way, all the sodium piping was cut off using the cutting bits.

The cover gas pressure in the secondary argon gas system at normal operation is controlled in the range of 34 to 44 kPa. It was decided to lower the cover gas pressure to 220 Pa at for the cutting work based on the experience of taking the surveillance materials out of the system, and the glove box pressure ratings. An automatic low-pressure argon control system was newly installed in the system. The oxygen concentration inside the glove box was controlled below the level of 1000 ppm. The control value of oxygen and nitrogen concentration in the system sodium piping is below 300 and 1200 ppm, respectively.

10.3. Future schedule

The performance of newly installed components and cooling systems will be confirmed through a series of functional system tests. The core replacement will start in the summer of 2002. When the initial MK-III core configuration is complete, the reactor power will be increased in steps to conduct performance tests that confirm the core physics and plant characteristics. The MK-III rated power operation will start in 2003.

11. CONCLUSIONS

The successful operations of JOYO provide a wealth of experiences with core management, impurity control, reactor engineering tests, innovative instrumentation techniques, operation and maintenance support systems, and component modifications. These experiences and accumulated data are to be used for the design of future fast reactors. They are also useful for upgrading the JOYO core and plant to the MK-III configuration and are essential to secure steady and safe reactor operation and enhance the irradiation capability of JOYO in the future.

REFERENCES

[1] MAEDA, S., et al., Fast Reactor Core Management in Japan: Twenty Years Evolution at JOYO, Proc. 5th International Topical Meeting on Research Reactor Fuel Management (RRFM 2001) Eurogress Aachen, Germany, 1-4 April 2001, European Nuclear Society (ENS) (2001) pp. 56-60.

[2] TABUCHI, S., AOYAMA, T., Development of JOYO MK-II Core Characteristics Database, Proc. Symp. on Nuclear Data, Ibaraki, Japan, 18-19 November 1999, Naoki Yamano and Tokio Fukahori (Eds), JAERI-Conf 2001-006 (2000) pp. 149-153.

[3] TAKANO, H., KANEKO, K., Revision of Fast Reactor Group Constant Set JFS-3-J2, JAERI-M 89-141 (1989).

[4] NAKAGAWA, T., et al., Japanese Evaluated Nuclear Data Library Version 3 Revision-2: JENDL-3.2, J. Nucl. Sci. Technol., **32**, 12 (1995) p. 1259.

[5] TAKAMATSU, M., AOYAMA, T., Neutron Intensity of Fast Reactor Spent Fuel, Proc. Symp. on Nuclear Data, Ibaraki, Japan, 27-28 November 1997, Tadasi Yoshida and Tokio Fukahori (Eds), JAERI-Conf 99-002 (1998) pp. 21-26.

[6] Atom Percent Fission in Uranium and Plutonium Fuel (Neodymium) Annual Books of ASTM Standard, E-321-79 (1971).

[7] MAEDA, S., AOYAMA, T., Decay Heat of Fast Reactor Spent Fuel, Proc. Int. Conf. on Nuclear Data (ND2001), Ibaraki, Japan, 7-12 October 2001, JAERI, published as Journal of Nuclear Science and Technology, Supplement 2 (2002).

[8] TASAKA, K., et al., JNDC Nuclear Data Library of Fission Products -Second Version-, JAERI 1320 (1990).

[9] SAWADA, M., et al., Experiment and Analysis on Natural Convection Characteristics in the Experimental Fast Reactor JOYO, Nuclear Engineering and Design, **120** (1990) pp. 341-347.

[10] AOYAMA, T., et al., The Operational Experience of the Experimental Fast Reactor JOYO, Proc. 3rd Asian Symposium on Research Reactor, Ibaraki, Japan, 11-14 November 1991, JAERI (1991) pp. 75-82.

[11] AOYAMA, T., ITO, C., Integral Test on Activation Cross Section of Tag Gas Nuclides Using Fast Neutron Spectrum Fields, Proc. Int. Conf. on Nuclear Data (ND2001), Ibaraki, Japan, 7-12 October 2001, JAERI, published as Journal of Nuclear Science and Technology, Supplement 2 (2002).

[12] WATANABE, K., et al., Development of Failed Fuel Detection and Location Technique Using Resonance Ionization Mass Spectrometry, J. Nucl. Sci. Technol., **38**, 10 (2001) pp. 844-849.

[13] IIZAWA, K., et al., Transport of Radioactive Corrosion Product in Primary Systems of a Sodium Cooled Fast Reactor, Proc. Material Behavior and Physical Chemistry in Liquid Metal Systems 2, Plenum Press, ISBN 0-306-45069-0 (1995) pp. 9-26.

[14] AOYAMA, T., et al., Application of Optical Fiber for Radiation Measurement in Fast Reactor Primary Cooling System, Proc. 12th Pacific Basin Nuclear Conference, Seoul, Korea, 29 October – 2 November 2000, Vol. 2, Korea Atomic Industrial Forum and Korea Nuclear Society (KAIF-KNS) (2000) pp. 095-1105.

[15] HIMENO, Y., et al., Improvement of Man-Machine Interaction by Artificial Intelligence for Advanced Reactors, Reliability Engineering and System Safety, **38** (1992) pp. 135-144.

[16] SAWADA, M., et al., Operation and Maintenance Support Systems of the Experimental Fast Reactor JOYO, Proc. Int. Topical Meeting on Safety of Operating Reactors, Seattle, USA, 17-20 September 1995, ANS (1995) 564-570.

[17] UEDA, M., et al., Condition Monitoring System of Rotating Machines in the Experimental Fast Reactor JOYO, Proc. 3rd JSME/ASME Joint Int. Conf. on Nuclear Engineering (ICONE-3) Kyoto, Japan, 23-27 April 1995, Vol. 3, ASME (1995) pp. 1619-1621.

[18] MAEDA, Y., et al., Current Status and Upgrading Program of the Experimental Fast Reactor JOYO, Proc. World Nuclear Congress and Expo (ENC'98) Trans., Nice, France, 25-28 October 1998, Vol. IV, Workshops, (1998) pp. 21-30.

[19] SUZUKI, S., et al., Upgrading Program of the Experimental Fast Reactor 'JOYO' - The MK-III Program -, Proc. 10th Pacific Basin Nuclear Conference, Kobe, Japan, 20-25 October 1996, Vol. 1, Atomic Energy Society of Japan (AESJ) (1996) pp. 759–768.

[20] YOSHIDA, A., et al., Upgrading Program of the Experimental Fast Reactor Joyo, Book of Abstracts Int. Conf. on Nuclear Engineering (ICONE-9), Nice, France, 8-12 April 2001, Vol. 2, ASME (2001) p. 492.

OPERATIONAL EXPERIENCE OF BN-350

I. DUMCHEV
MANGISHLAC ATOMIC ENERGY COMPLEX (MAEC), KAZAKHSTAN

Abstract

The BN-350 is a fast breeder reactor located in Aktau, Kazakstan on the eastern shore of the Caspian Sea. The reactor began operation in 1972. It operated on an open uranium fuel cycle optimized to produce plutonium for the USSR weapons complex and generated steam for electricity, heat and seawater desalination. The reactor was in operation until April 1999 when decision for final shutdown had been taken by Government. In this report the operational experience is considered for the following equipment that directly contacts the liquid metal coolant:

- The main circulation pumps of primary circuit;
- Leakage hold-up tank for the main circulation pump of loop No 4;
- Non-return valves;
- Steam generator superheaters for design steam generators;
- Cold filter-trap of the primary circuit purification system;
- Drum for spent fuel assemblies.

All the above listed equipment is selected from the point of view of occurrence the technical problems during both the commissioning works and operation of the equipment. The ways are indicated in the report how these problems have been solved.

1. THE MAIN CIRCULATION PUMP OF PRIMARY CIRCUIT

The main circulation pump (pump) is intended for providing the sodium circulation in primary circuit. The pump is centrifugal, vertical arrangement, single-stage, console, sealed, electrically driven type. Motor — asynchronous, double-speed, vertical arrangement with squirrel-cage rotor.

TABLE 1. SPECIFICATION OF THE MA IN CIRCULATION PUMP OF PRIMARY CIRCUIT

Item No	Designation of the value measured	Dimension	Numerical value
1	Capacity (rated)	m^3/h	3320
2	Head pressure (nominal)	kg/cm^2	1.4
3	Minimum permissible pressure at the pump inlet at nominal consumption	kg/cm^2	0.6
4	Transported liquid	Sodium	
5	The temperature of pumped liquid in operational condition	°C	300
6	Shaft number of revolutions in a basic speed mode in a low speed mode	 rpm rpm	 1000 250
7	The motor rated shaft power at 1000 rpm at 250 rpm	 KW KW	 55 55
8	Supply net voltage of the inducing winding secondary winding	 V V	 6300 380
9	Frequency of supply network	Hz	50
10	Excessive gas pressure in the pump gas chamber	$Kilogauss/cm^2$	0.9
11	Direction of the shaft rotation from the pump side	Counter Clock- wise direction	
12	Weight of removable part	Ton	18

1.1. Event description

1.1.1. Date

1973.

1.1.2. Reactor status

Commissioning works.

1.1.3. Fault

Vibration and increase of sodium level in the pump tank during the pump switching on to 1000 rpm. After relaxation of transitional processes the sodium level was stabilized and set within the permissible limits.

1.1.4. The way of fault detection

Indications of sodium level gauge in the pump tank; actuation of warning alarm system.

1.1.5. Defect origin

Defect of the project.

1.1.6. Equipment recovery, repair, or interchange

To eliminate vibration and increase of sodium level in the pump tank during the pump switching on to 1000 rpm the additional labyrinth seal was set between the removable part and pump body (Figs 1 and 2). The sodium level in the pump tank has become stable during the pump switching on to 1000 rpm.

2. THE PUMP LEAKAGE DRAIN HOLD-UP TANK

The pump leakage drain tank fulfills a function of both the gas separator and hydraulic seal that prevents the gas entrainment into suction pipeline of the main circulation pump. The pump leakage drain tank consists of the tank and removable part coupled by the cap flange, bolts and nuts. The coupling tightness is achieved by welding. Deflector improves separation of gas by changing the sodium flow direction. The pressure compensated valve is the double-inlet for sodium. Pressure variation at the pump suction side and outlet nozzles makes no influence on the needle position and sodium consumption via the pump leakage drain tank. The sodium flows out of the pump tank into the pump leakage drain tank via the pipeline of 219 mm outside diameter and 12 mm wall thickness. From the pump leakage drain tank the sodium flows via the pipeline of 159 mm diameter and 6 mm wall thickness to DU500 pipeline into the suction pipeline of circulation pump.

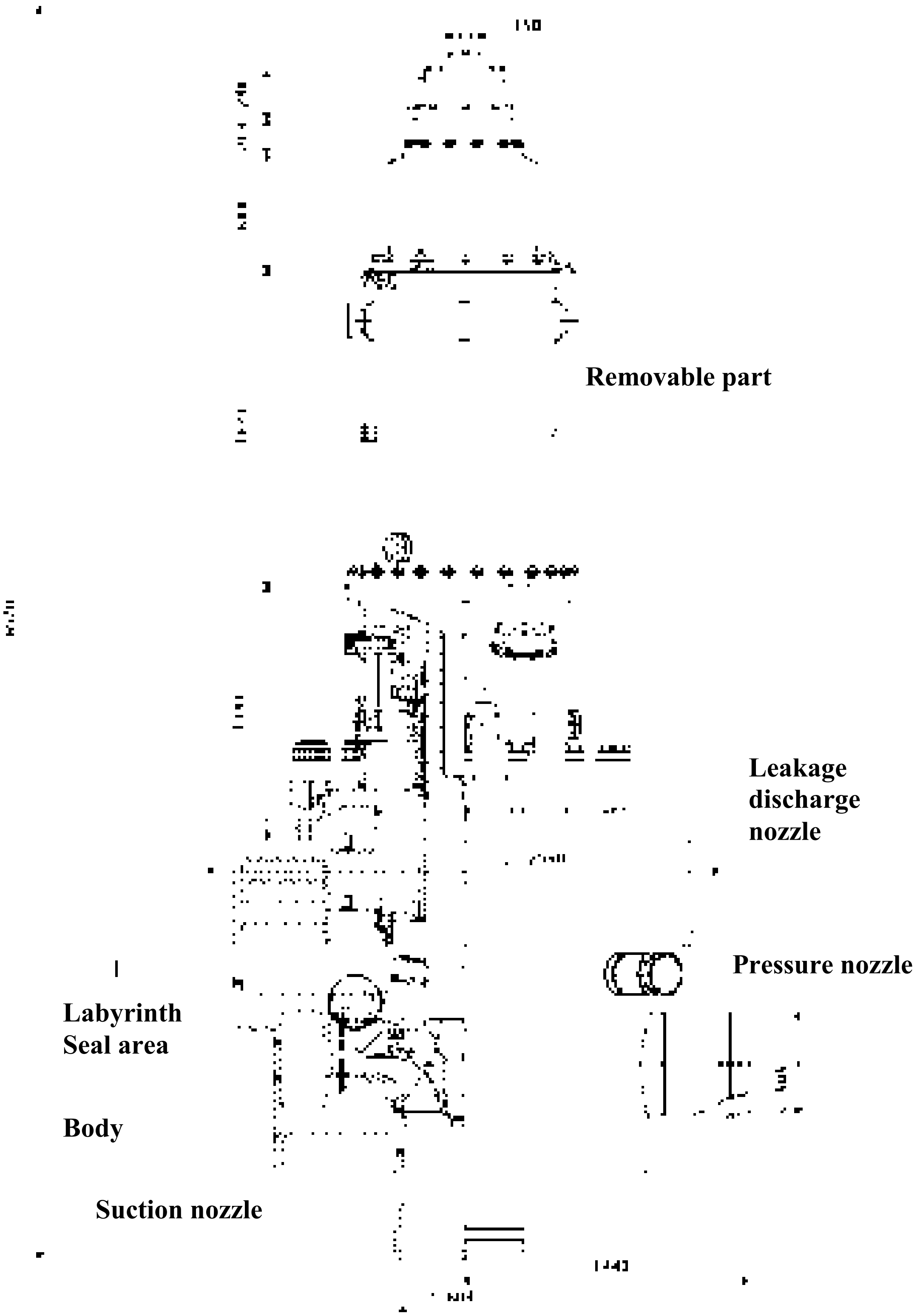


FIG. 1. The main circulation pump of primary circuit.

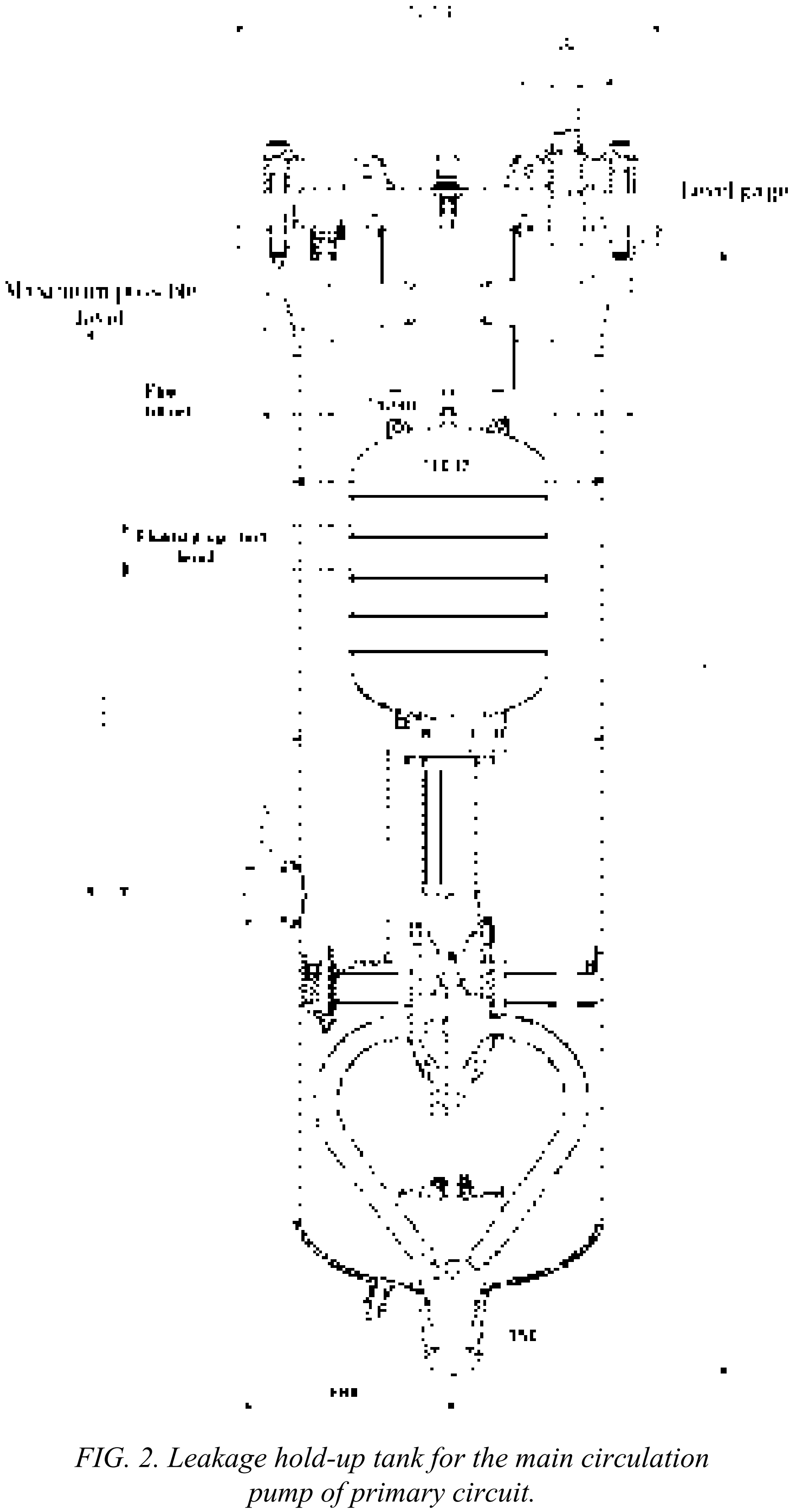


FIG. 2. Leakage hold-up tank for the main circulation pump of primary circuit.

2.1. Event description

2.1.1. Date

1977-1992.

2.1.1.2. Reactor status

Reactor operation at 650÷750 MW power.

2.1.1.3. Fault

Self-induced vibrations of sodium level in the pump leakage drain tank during operation of the main circulation pump (pump) at 1000 rpm in the loop No 4 of primary circuit. The self-induced vibrations eliminated by the cutting off the loop No 4 via the gas communications from primary circuit and by maintaining the gas pressure in the loop No 4 higher than that in all the rest primary circuit gas volume.

2.1.1.4. The way of fault detection

Indications of both the sodium level gauge in the pump tank and “Signal” float position alarm.

2.1.1.5. Defect origin

The hangers’ spring tension had changed in the pipe that connects the pump tank and leakage drain tank during operation. This resulted in a change of the pipelines’ angle of slope and caused the pipeline “sagging”. In this area the whole cross-section of pipeline was filled with sodium and there was a temporary reduction in sodium draining to the leakage drain tank. Therefore the level of leakage drain tank fell. Then the sodium level raised in the pump tank due to reduction of sodium draining. After raising of level in the pump tank the sodium drained from stagnant area to the leakage tank and thus the level in the leakage hold-up tank started raising. After drop of sodium level in the pump leakage drain tank the process repeated causing the period of self-induced vibrations 1.6 second (Fig. 3).

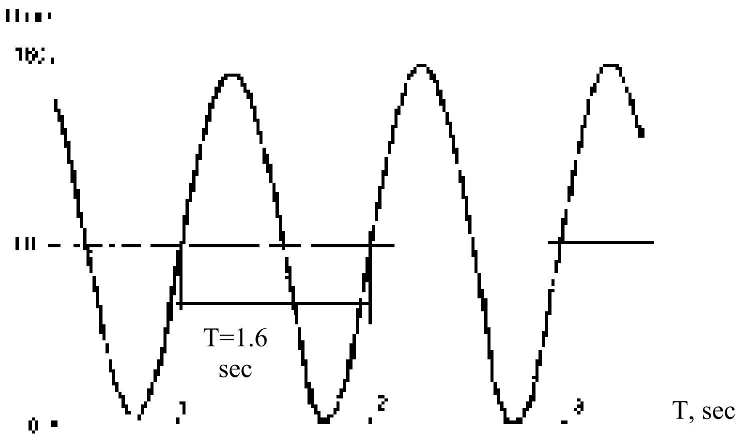


FIG. 3. Self-induced vibrations of the leakage drain tank float (indications of the “Signal” alarm indicator).

2.1.1.6. Equipment recovery, repair or interchange

It was very complicated problem to find out the exact reason for self-induced vibrations of sodium level in the leakage drain tank because of the discrete indications of the pump tank sodium level indicator and the smaller sodium level vibration amplitude than the indicator scale interval. So the indicating instrument showed the conservative value for sodium level in the leakage drain hold-up tank. The first solution was to change the leakage tank float (with removable part). The change of removable part did not stop the self-induced vibrations in the leakage tank. After that the adjustment was carried out of the hangers' spring tension for the pipe connecting the pump tank and pump leakage drain tank. This way the self-induced vibrations of sodium level had been stopped.

3. THE NON-RETURN VALVE OF PRIMARY CIRCUIT LOOP

3.1. Design specification

The non-return valve is intended to prevent the back flow of coolant in the loop when the pump is tripped while other loops operate. The non-return valve consists of the body and the removable part (ass. 350E), as can be seen in Fig. 4.

The non-return valve body is a cylinder with two nozzles of diametrically opposite arrangement. The nozzles are welded to the lower part of the body. The inlet nozzle is located 143 cm higher than the outlet one. The internal edge of inlet nozzle is cut at a 13° angle relative to the vertical axis from the up-wise to down-wise direction. Prescribed service life of the non-return valve — 30 years. In accordance with the results of investigations the 200 000 hours service life was prescribed for the valve removable part (assemblage 350E), see Fig. 5.

Technical specification of the non-return valve (including the removable part — assemblage 350 E can be seen in Table 2.

TABLE 2. TECHNICAL SPECIFICATION OF THE NON-RETURN VALVE (INCLUDING THE REMOVABLE PART — ASSEMBLAGE 350 E)

	The pump operational model	Parameters	Unit of means	Numerical value
1	The trip of one operating pump	Maximum sodium back consumption	m^3/h	650
		Pressure jump	kg/cm^2	5-6
		Duration of sodium back consumption	S	0.45
		Return valve close time	S	2.65
2	The first pump startup $P_{gas} = 0.6$ atm	Maximum back consumption	m^3/h	110
3	The first pump startup to 1000 rpm, $P_{gas} = 0.9$ atm	Maximum back consumption	m^3/h	110

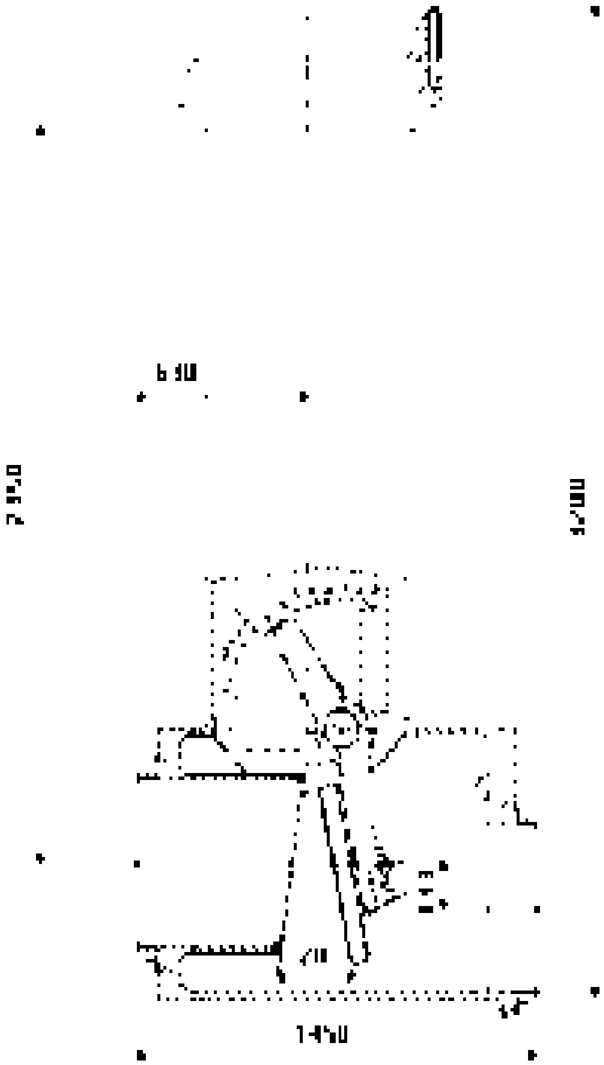


FIG. 4. The design non-return valve (ass.350).

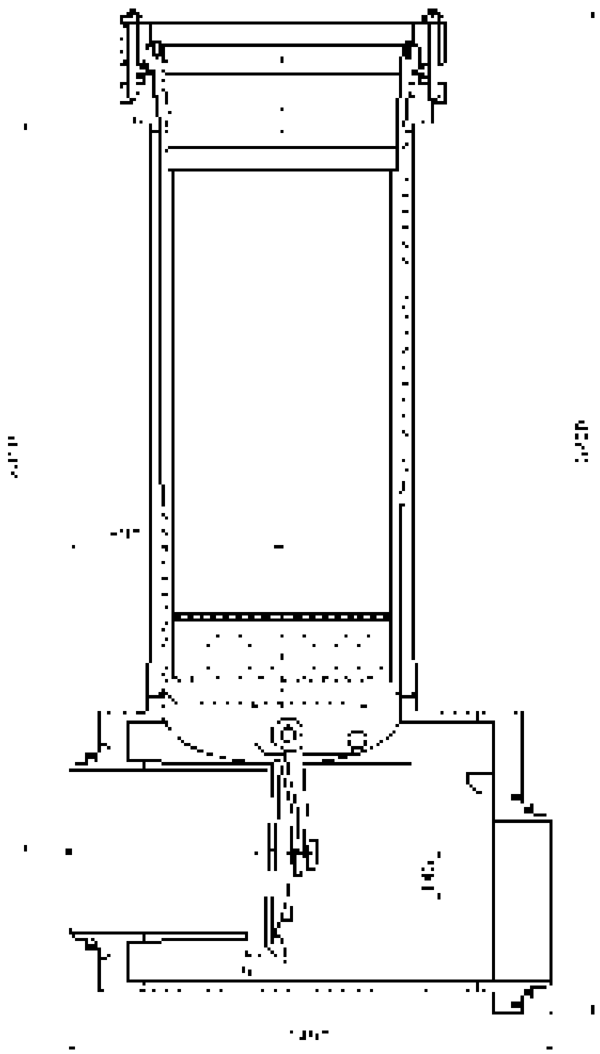


FIG. 5. Updated non-return valve (ass. 350E).

3.2. Event description

3.2.1. Date

1973.

3.2.2. Reactor status

Commissioning works.

3.2.3. Fault

The design return valve (including the removable part, ass. 350) in the mode of the trip for one of the five pumps operating at 1000 rpm cuts off the loop, which is to be eliminated from operation with the coolant back consumption ≈ 2000 m^3/h. The non-return valve closing was accompanied by the valve disk - seat collision followed by hydraulic shock that caused the vibration of cutoff loop pipelines in primary circuit.

3.2.4. The way of fault detection

Visually observed shifting the primary circuit pipelines.

3.2.5. Defect origin

Defect of the project.

3.2.6. Total radioactivity release

No.

3.2.7. Equipment recovery, repair or interchange

In accordance with test results the design part is changed for updated removable part (assemblage 350 E). A few valve versions supplied with different removable parts (ass. 350, 350 D, 350 B, 350) were designed and tested.

3.2.8. Measures on avoidance the event recurrence

The tests have been conducted for updated versions of valves in startup and trip modes of pump. The optimum design version of the valve has been chosen.

4. STEAM SUPERHEATER

4.1. Technical specification

The superheater (Fig. 6) is intended for superheating the dry saturated steam incoming from evaporator up to 435°C temperature. The superheater is a vertical U-shaped vessel with the upper arrangement of inlet and outlet chambers for sodium and steam. The superheater body is manufactured from 778 mm diameter cylindrical shells of 24 mm wall thickness for sodium chamber and 33 mm for steam chamber.

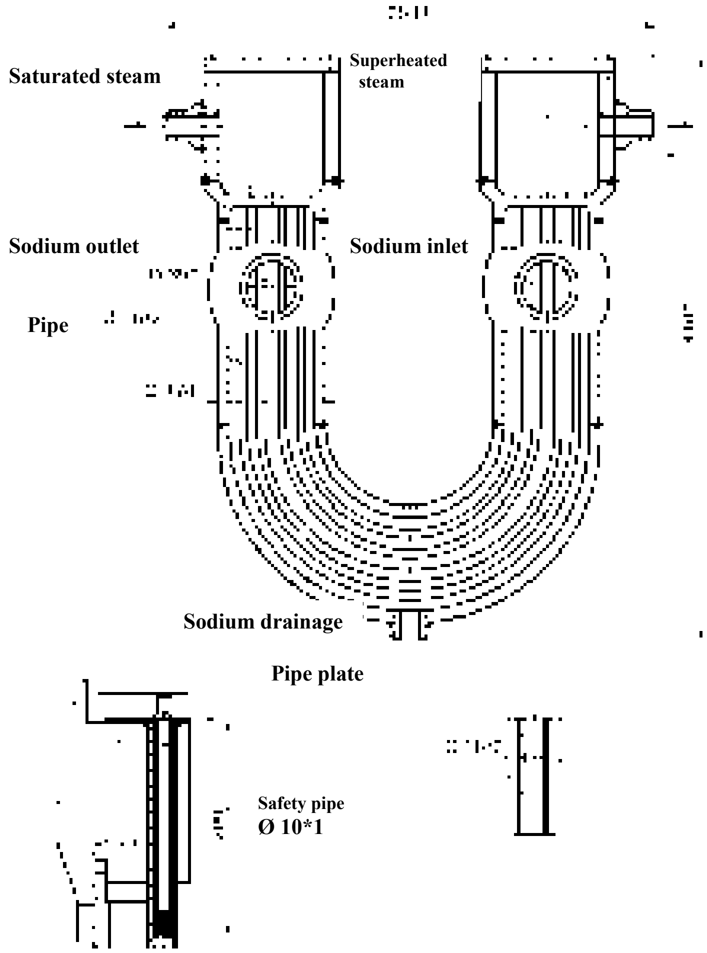


FIG. 6. The design of superheater of steam generator.

Table 3 shows the steam superheater performance specification (design).

TABLE 3. THE STEAM SUPERHEATER PERFORMANCE SPECIFICATION (DESIGN)

coolant pressure	6 bar
coolant inlet temperature	453°C
coolant outlet temperature	419°C
steam pressure	50 bar
maximum permissible pressure before opening the safety valve	54-56 bar
inlet steam temperature	265°C
outlet steam temperature	435°C
weight (dry)	12 200 kg
heating area	
Materials of the superheater components:	
body, piping plates and piping bundle	steel 1Cr2Mo
saturated and superheated steam chambers	steel 22 K

The evaporator saturated steam chambers are interconnected by ∅273×11 diameter pipes. The evaporator and superheater fastening is conducted with an account of the piping temperature expansion. Each evaporator and superheater is supplied with one movable and one immovable support. The prescribed service life of design SG superheaters is 30 years.

4.2 Event description

4.2.1. Date

1973-1974.

4.2.2. Equipment status

Commissioning works after overhaul of steam generators.

4.2.3. Fault

Increased vibration (up to 3g) of the superheater heat-exchange pipes located mainly along the pipe bundle periphery.

4.2.4. The way of fault detection

Vibration measurements during commissioning works.

4.2.5. Defect origin

Defect of the project.

4.2.6. Equipment recovery, repair, or interchange

During SG (evaporators) overhaul in pursuance with the results of vibration investigations the pipes were plugged that were operated at maximum vibration acceleration amplitudes ≤3g.

4.2.7. Measures on avoidance the event recurrence

After plugging the pipes measurements on vibration of the superheater piping were carried out in every 10 000 hours of operation. In 1979 the destructive method investigations of superheaters and SG No 4 were carried out using destructive method (after change of the design SG No 4 for a steam generator manufactured in Czechoslovakia). The investigations conducted have shown zero vibration wearing of superheater pipes. In accordance with the results of investigations a decision was made to conduct the vibration measurements for piping in 2000 hours.

5. THE PRIMARY CIRCUIT SODIUM PURIFICATION SYSTEM

The system for sodium purification in the cold filter-traps is intended for purification of primary circuit sodium from oxides, hydrates, carbonates and other impurities. The sodium purification system comprises five cold filter-traps with connection to charge and discharge pipelines Purification system is connected to loops No 2 and 3 of primary circuit. The sodium purification scheme can be seen in Fig. 7.

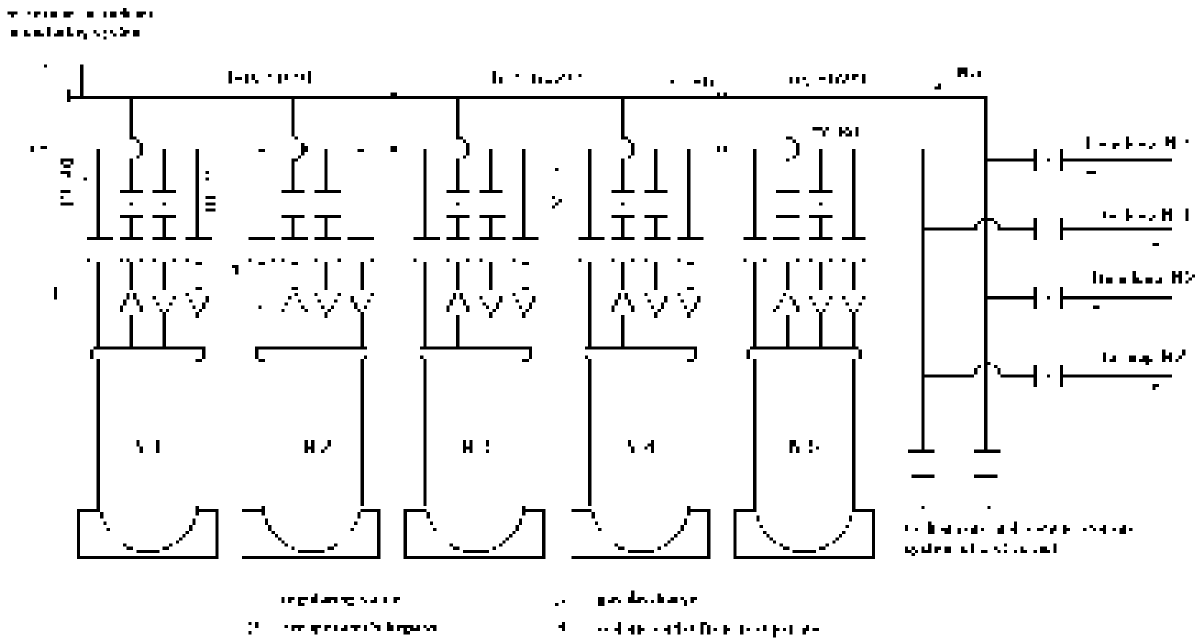


FIG. 7. Sodium purification system.

Sodium flows out of the pressure head leg of DU500 main (second or third loop) to the pressure header of primary circuit sodium purification system and then flows to operable cold filter-trap. Here sodium is purified from soluble impurities owing to its cooling to the impurity setting-out and trapping temperature. After purification sodium is returned to (second or third loop) to the pump suction pipeline.

5.1. Cold filter-trap

The trap is intended for trapping the oxides, hydrates, carbonated and other impurities from the molten sodium.

5.2. Event description

5.2.1. Date

January 1982. Sodium leak in primary circuit along the “sphere-to-cone” coupling recuperator bypass-cold filter-trap No 2 after heating the frozen part of thermal area up to 170°C.

5.2.2. Reactor status

Heat power 700 MW.

5.2.3. Faulty system

The primary circuit sodium purification system.

5.2.4. The way of fault detection

Visual — smoke in central hall; actuation of falt signaling — increase of aerosol specific activity, reduction of insulation resistance in the upper electric heating area of cold filter-trap (see Fig. 8).

5.2.5. Failured equipment

Bypass pipe connector “sphere-to-cone” of recuperator, the upper thermal area of cold filter-trap electric heating (see Fig. 9).

5.2.6. Defect origin

Hydraulic shock after occurrence of consumption via the trap resulted from heating thermal area; impact of temperature gradient with the micro-leakage available; defect of coupling assemblage “sphere-to-cone”.

5.2.7. Total radioactivity release

0.03 Ci.

5.2.8. Equipment recovery, repair or interchange

Defective connector “sphere-to-cone” is cut and replaced for pipe.

5.2.9. Measures on avoidance the event recurrence

All the design pipeline couplings “sphere-to-cone” are changed; procedures are adjusted when putting the cold filter-traps into operation.

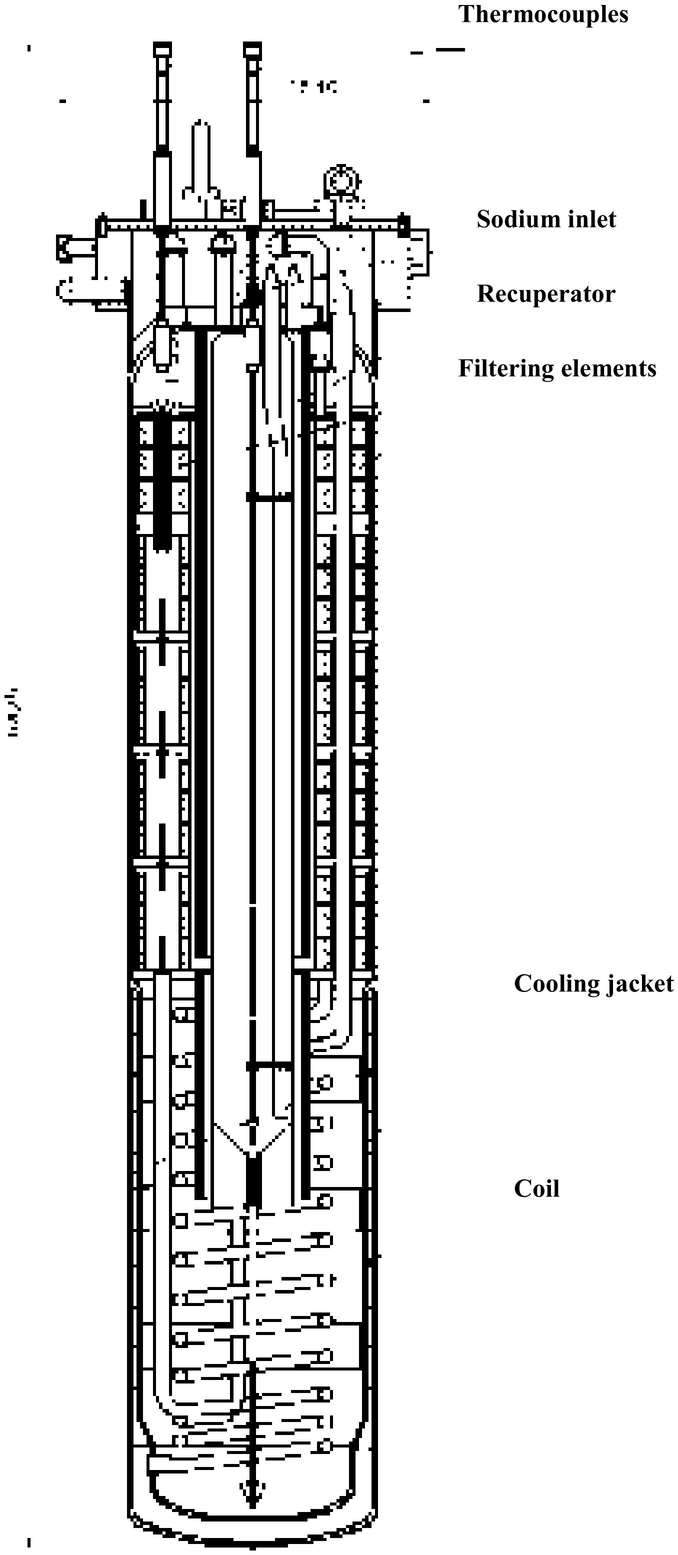


FIG. 8. Cold filter-trap.

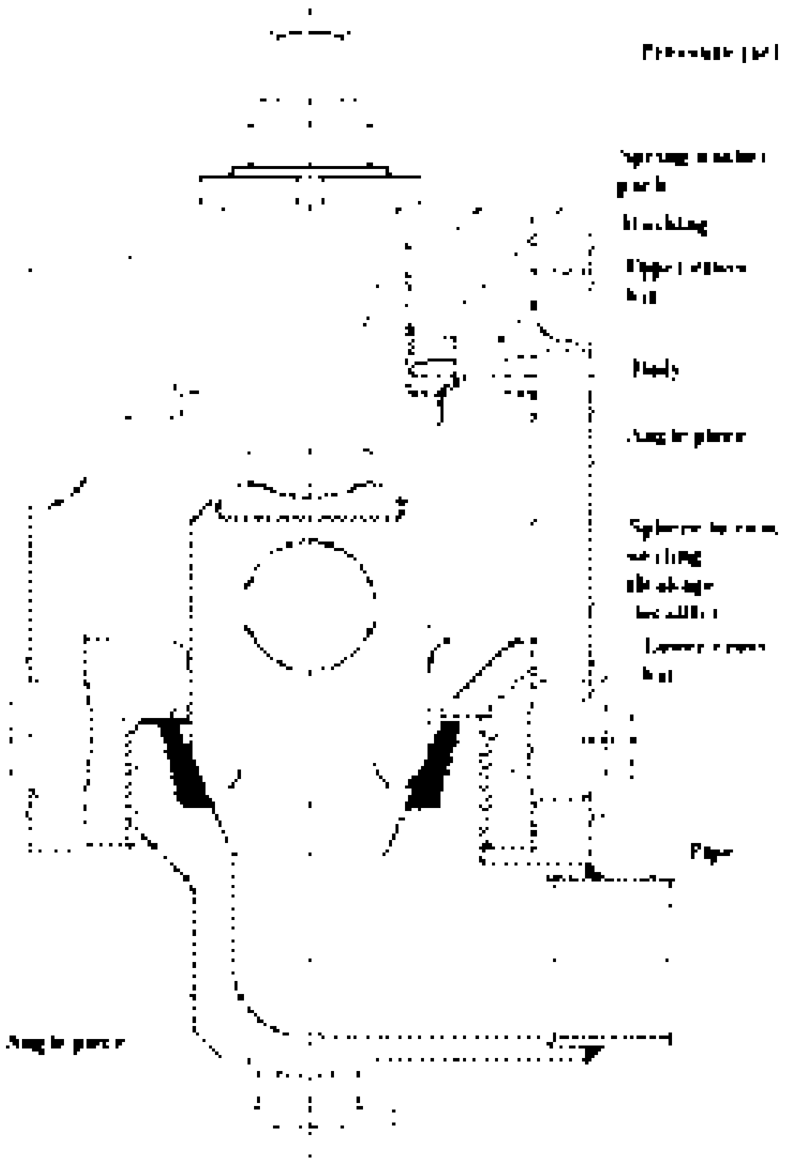


FIG. 9. Sphare-to-cone-connection.

6. COOLING SYSTEM FOR SPENT ASSEMBLIES

6.1. Brief description of the facility

The facility is an interim storage facility and is intended for interim accommodation and cooling of spent fuel assemblies (SFA) unloaded from the apparatus. The facility consists of a bath, drum for spent assemblies (DSA), drum drive, the drum drive shaft, drum automatic control mechanism, drum truck, displacer truck, displacer, truck drive, drive counterweight, dosimetric chamber cells, bellows compensator and screens. There are 38 cells for SFAs in the drum (see Fig. 10). The bath is supplied with air ducts for the cooling air inlet and outlet (see Figs 11 and 12). The following brunch-pipes are located in the upper part of the bath: Loading one — connects the bath and transfer box and is intended for SFA loading into the drum cells. Unloading one — connects the bath and washing box and is intended for SFA unloading from the drum cells.

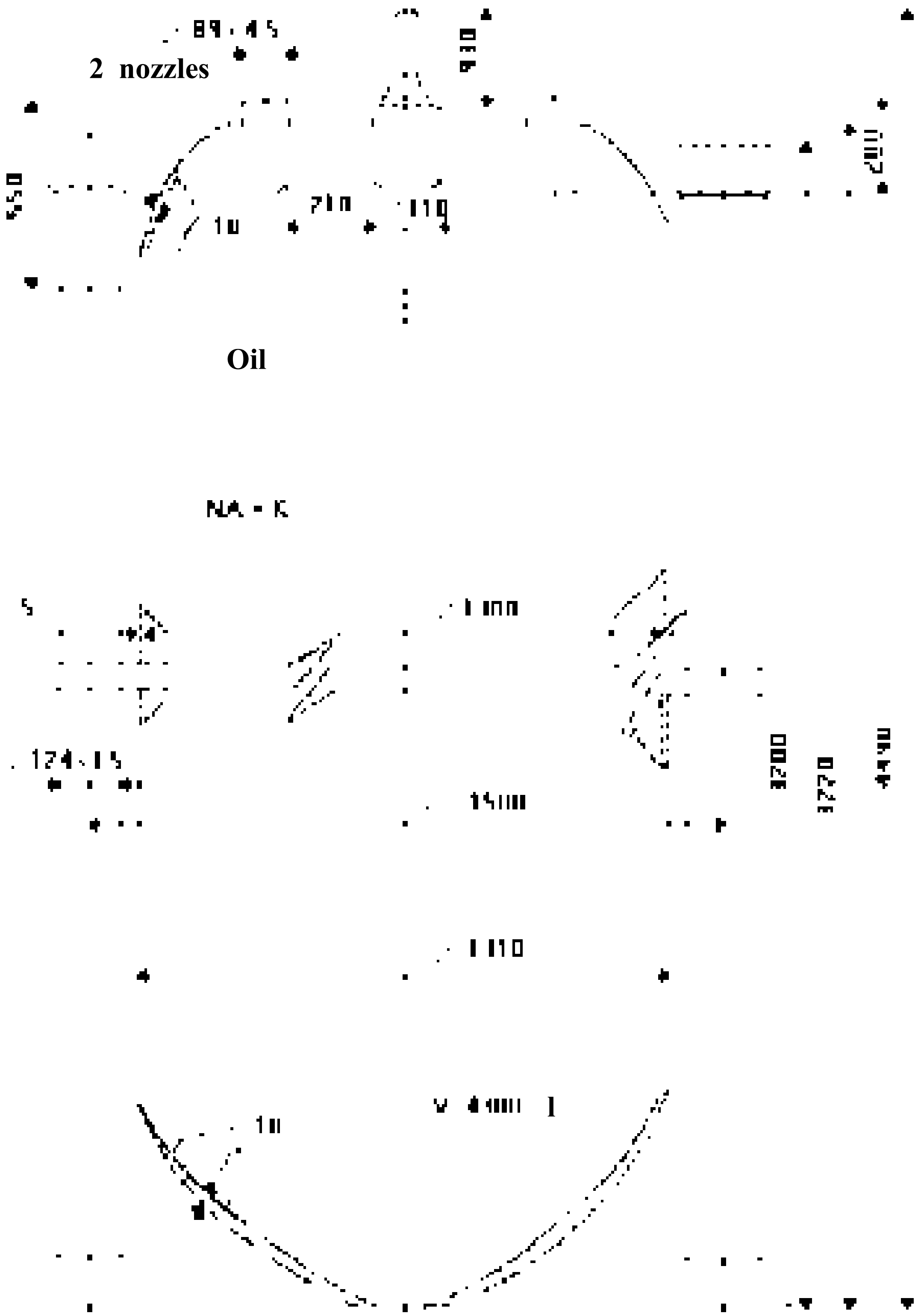


FIG. 10. FAs drum.

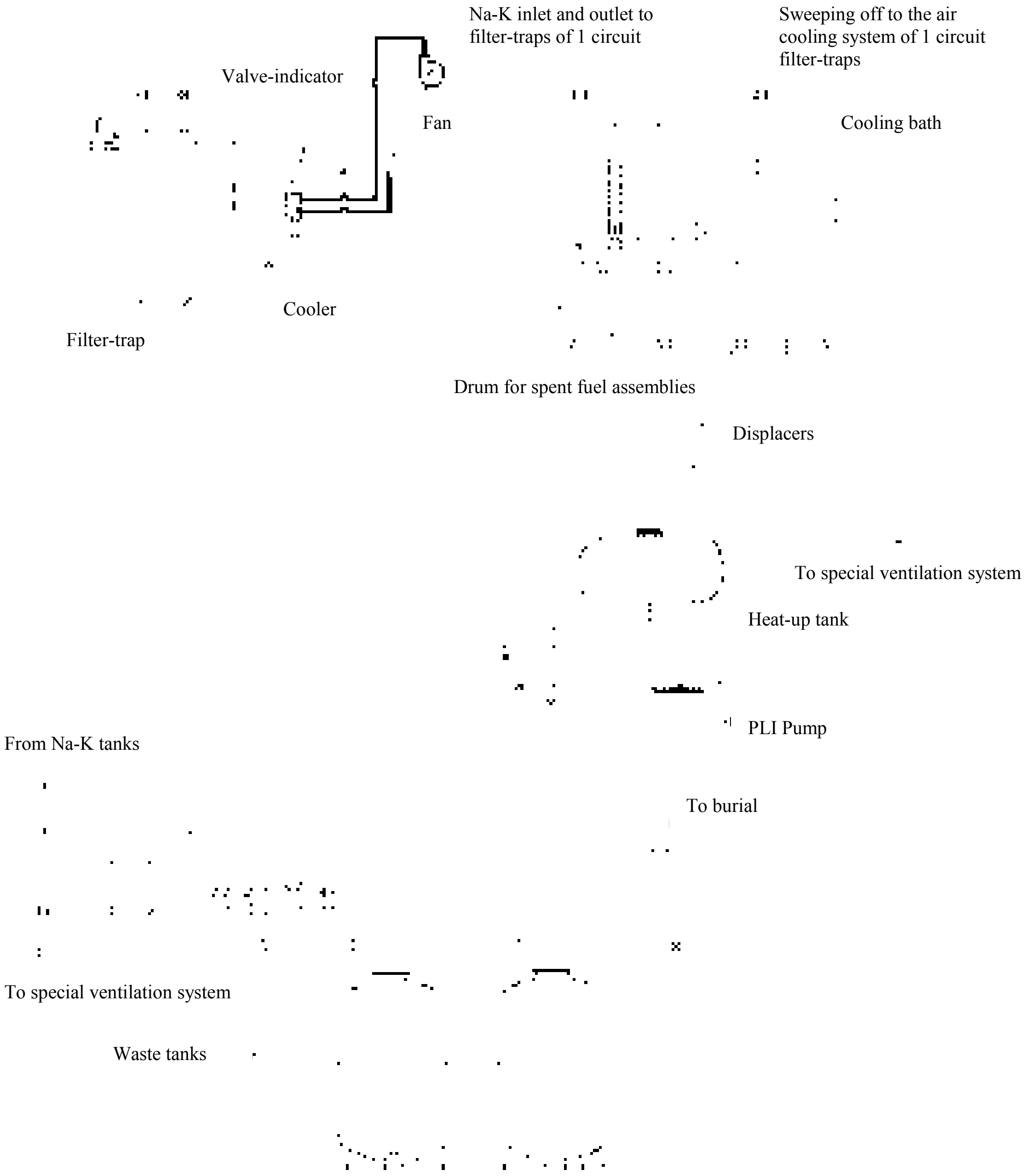


FIG. 11. Cooling system for spent fuel assemblies.

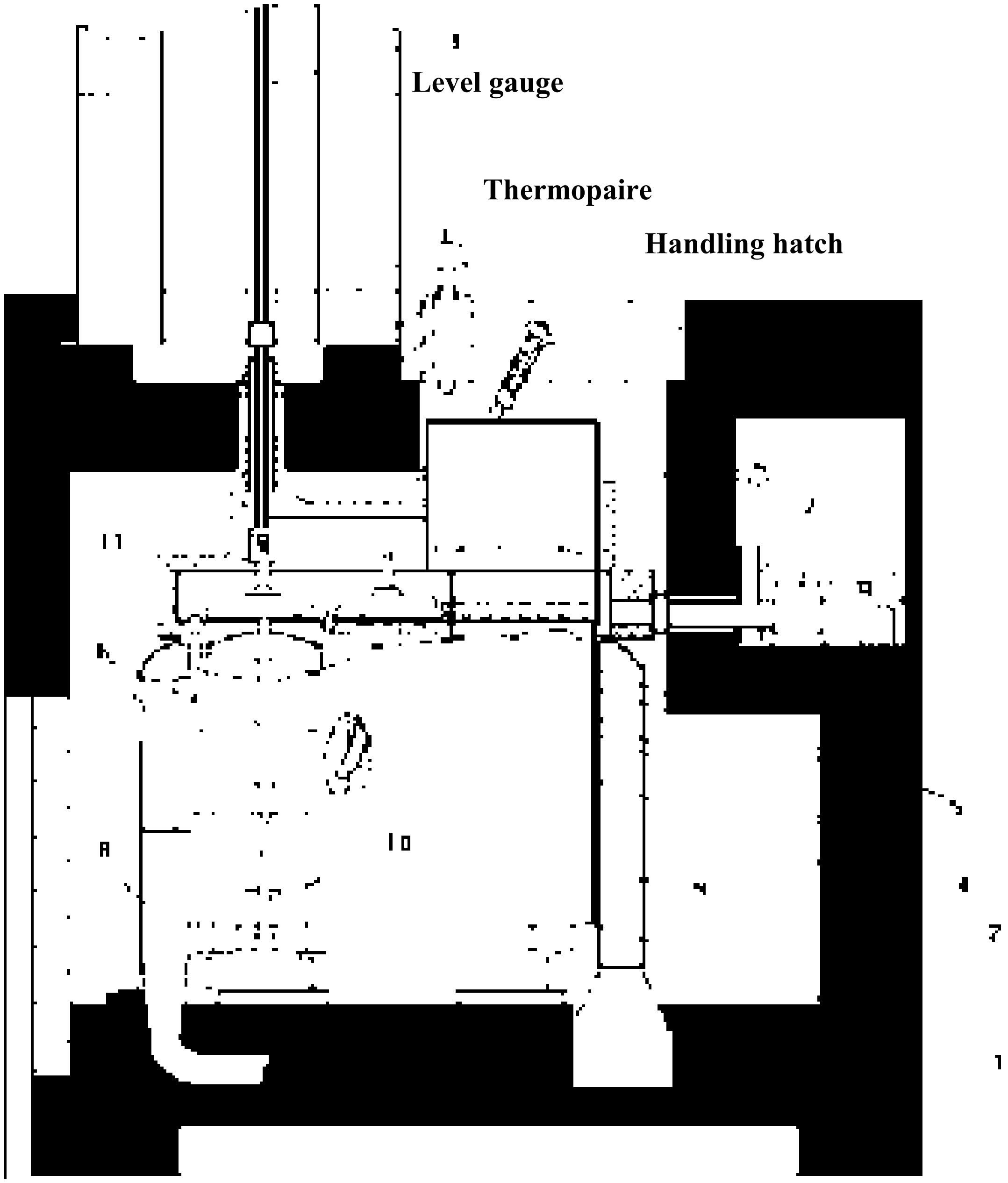


FIG. 12. Facilities arrangement of cooling system.

1 – Bath
2 - Drum
3 - Driving spindle
4 - Drum and displacers carriage
5 - Displacers
6 – Shields
7 - Carriage drive and counterbalancer
8 - Air outlet
9 - Air inlet
10 - Socket for camera installation
11 - Bellows compensators

6.2. Event description

6.2.1. Event

In April of 1976 when preparing to SFAs drum unloading (36 SFAs from interim storage facility) followed by SFAs washing the lack of drum rotation was detected.

6.2.2. Reactor status

Reactor 650 MW heat power operation.

6.2.3. Failed system

SFAs cooling system.

6.2.4. The way of fault detection

Visually inspection revealed the immovability of the drum remote and manual drive.

6.2.5. Failured equipment

Mechanical damage of the drum drive bearing assembly during removal of 36 SFAs from the drum.

6.2.6. Defect origin

The moisture ingress from washing box followed by formation of solid sodium peroxides; staff error.

6.2.7. Total radioactivity release

No.

6.2.8. Equipment recovery, repair or interchange

After draining the Na-K alloy into the drain tanks the water-oil washing of cooling bath was conducted, all the SFAs were removed from the drum, the drum was removed from cooling bath and placed to the well equipped storage area of reactor building with maintenance of inert gas medium inside the dram. Some two barrels were fitted into the cooling bath (one in refueling box another in washing box) for placing the SFAs, fuel rods, control rods and sleeves of control and protection system (CPS).

6.2.9. Measures on avoidance the event recurrence

Decision was made to give up the usage of interim SFAs storage facility. The washing operations were conducted simultaneously with refueling ones. The emergency reloading system (ERS) that uses inert gas for FAs cooling became the standard one for replacement of SFAs, control rods and sleeves (from reloading box into the washing one). The SFAs cooling system was mothballed.

LIFETIME EXTENSION OF THE PHÉNIX NUCLEAR POWER PLANT

L. MARTIN, D. PEPE
CEA–Phénix, Bagnols-Sur-Cèze Cedex, France

R. DUPRAZ
Framatome-ANP, Lyon Cedex, France

Abstract

The French fast reactor prototype: Phénix, located at Marcoule in the Gard department, was put into commercial operation in 1974. The total time of power operation of the plant is approximately 100 000 hours. The initial objective of Fast Breeder Reactor demonstration has been achieved. Since the mid-nineties, the role of the reactor as an irradiation facility has been emphasized, particularly in support of the CEA's transmutation R&D programme in the context of the 30th December 1991 French law on long-lived radioactive waste management. This new objective has required the extension of the planned reactor lifetime. A renovation programme was defined based on:

- Plant safety improvements based on today's standards;
- Evaluations and inspections of components in order to assess their condition and to identify possible damaging mechanisms;
- The estimation of the ability of components to continue operation taking into account the experimental feedback.

The safety upgrading of the plant consisted essentially of the following:

- The addition of a safety control rod to the reactor. The partitioning of the secondary sodium circuits in the Steam Generator building to improve protection against sodium fires;
- The installation of an anti- whip system on the high-pressure steam pipes;
- The construction of two redundant seismic resistant emergency water-cooling circuits;
- The seismic reinforcement of the plant buildings.

An extensive evaluation campaign was carried out on all the equipment that is essential to safety. Damaging phenomena were systematically investigated. Following the evaluation, the equipment was either repaired or replaced, or it was demonstrated that equipment condition allowed lifetime to be extended. Among the major works performed was the replacement of the 321 stainless steel hot leg of the secondary loops and the repair of the superheater and reheater SG modules. This latter operation is currently underway. The repair process consists in dismounting the modules, washing the internal residual secondary sodium, replacing potentially defective parts and remounting. Special analysis was conducted regarding the reactor block structures. The past and future damage rates were evaluated, involving investigation of manufacturing data, determination of the behaviour of some particular mechanical assemblies such as welded joints, estimation of the loadings from operating conditions associated with thermal calculations. In addition to these studies, and as a second defence line, the main reactor structures were inspected using various techniques. Because of the difficult access to the structures, original inspection procedures had to be developed and special equipment designed and manufactured.

The inspections included:

- Ultrasonic examination of the reactor vessel upper hangers;
- Ultrasonic examination of the core support conical shell;
- Televisual examination of the core cover plug;

Most of the renovation programme of the plant has been completed. The major remaining task is the repair of the SG modules. The power rising of the reactor is expected before the end of 2002. Six 120 EFPD operating cycles are scheduled to carry out the irradiation programme, representing about five-and-a-half years of operation. Considerable experience has been gained from the lifetime extension project in many fields, including the methodology that was developed, materials and components behaviour, inspection techniques.

1. THE ATOMIC ENERGY COMMISSION (CEA) OBJECTIVES

1.1. The French fast reactor demonstration prototype

PHÉNIX, located at Marcoule, in the Gard department, with a nominal 250 MWe power rating (563 MWth), was commissioned in 1974. It has currently provided approximately 100 000 hours of grid-connected operations, at operating temperatures of 560°C for the reactor hot structures.

The plant has achieved the objectives of demonstration of fast breeder reactor technology which were set at the time of construction, including the following significant examples:

- Average burnup increasing from 50 000 MWd/t to 100 000 MWd/t, with maximum burnup exceeding 150 000 MWd/t. These levels were reached with 8 cores of fuel which is 166 000 fuel pins.
- A measured breeding ratio of 1.16, which resulted in the loading of the first fuel assembly made with reprocessed plutonium in 1980.
- Gross thermal efficiency of 45.3%, allowing the production of over 20 billion kWh.
- Maintenance and operating ease due to a low dosimetry (overall dose to operators of 1790 mSv since start-up).

From 1992, the role of Phénix as an irradiation facility has been emphasized, particularly in support of the CEA R&D programme in the context of line 1 of the December 30th 1991 law on long-lived radioactive waste management. The first experiment, called SUPERFACT, led to the incineration of minor actinides (neptunium and americium). This programme was further strengthened in 1997, to compensate for the shutdown of Superphénix. It involves transmutation of Minor Actinides and Long-Lived Fission Products.

Since 1993, the reactor power has been limited to 350 MWth (145 MWe) on two secondary loop operations.

2. APPROACH, CONTEXT AND ORGANIZATION OF LIFETIME EXTENSION

The new objectives set by the Phénix power plant have led to an increase in operating life beyond the period originally planned at the time of start-up. In order to extend operations, the CEA has undertaken power plant renovation, based on the following:

- Safety improvement, taking into account changes in safety standards and current construction rules;
- Expert evaluations and inspections to examine component condition and state potential damage mechanisms;
- Assessment of the components' ability to continue operations, using feedback.

The studies started in 1993 and the renovation work in 1995. It was divided into two phases, separated by the 50^{th} cycle of operation in 1998. Extension of the operations was subject to the approval of the renovation programme by the Safety Authorities. The programme was reviewed in several steps by the Permanent Group in charge of the reactors.

Recommendations were issued by the Safety Authorities that were integrated in the programme.

For the renovation programme, CEA Nuclear Energy Direction has implemented specific organization different from plant operation organization. This organization involves several projects:

- A "lifetime extension" project, responsible for the theoretical research and experiments that evaluates component condition and component capacity to continue operating. This project also defines the preventive and curative maintenance actions and proposes operating improvements. This project is run by experts from CEA, EDF and Framatome-ANP;

- An on-site “Plant shut-down” project, responsible for co-ordination, planning of the works and logistics for contractors. It was itself divided into several projects;
- A “renovation” project, responsible for defining improvement and follow-through on attainment.

An independent group of earthquake experts, outside of the project team, was formed to validate the earthquake re-evaluation of the buildings. CEA Nuclear Energy Direction, assisted by EDF is the Contracting Authority for this work, working with an industrial organization made up of the Novatome Direction of Framatome-ANP as Prime Contractor and manufacturers:

- Specific projects responsible for defining and performing the reactor block inspections and Steam Generator repair;
- The “ten-yearly statutory maintenance” project, responsible for the 3rd ten-yearly inspection of all the elements subject to regulations.

3. SAFETY IMPROVEMENTS

Safety improvements emanate from the re-evaluation of installation safety in light of the current standards, primarily in the areas of earthquake resistance, spray sodium fires and water and steam pipe rupture. In order to guarantee the main safety functions with respect to reactor shutdown, to the decay heat removal and to the confinement of radioactive materials, in all the aforementioned accidental operating conditions, the following safety improvements were required:

- Installation of a new reactor shutdown system called SAC. This system is similar to the systems on the Superphénix reactor, and provides for complete mechanical decoupling between the absorber present in the core and the control mechanism located in the upper structures of the reactor block. A lifting electromagnet performs this decoupling.
- This system was placed in the reactor for the 50th operating cycle.
- Protection of the steam generator building against the consequences of a large spray sodium fire. The thermal effects of a fire were calculated based on the potential sodium leaks in the pipes in the steam generator building and were used to define the protections for the safety-related systems located in the building. The design codes used took into account the feedback from the tests performed on these types of fires at CEA Cadarache.
- Installation improvements consisted of creating 2 large cells to confine fire resulting from a sodium leak during a fire period of 30 minutes at a temperature of 1100°C, until the fire stops. The cell dimensions are 24 m high, 15 m wide and 10 m deep (Fig. 1).
- Protection of the steam generator building equipment from water/steam pipe rupture. Identification of the potential pipe break points and analysis of their consequences has led to the installation of specific equipment to protect against displacement. The 35-meter high, steel structure design of the steam generator building excludes the installation of the traditional anti-whip restraints or frames. An innovative design was developed made up of tie rods, pipe rings and shock absorbers that envelop the entire length of the pipe, thus restraining displacement (Fig. 2).
- Construction of 2 new redundant water circuits that replace the existing circuits and assist in the decay heat removal. Circuit design takes into account the consequences of an earthquake on the installation and of a large sodium fire on the secondary circuits. Each circuit includes a 1500-kW air cooler and 720 kVA stand-alone emergency electrical supply.

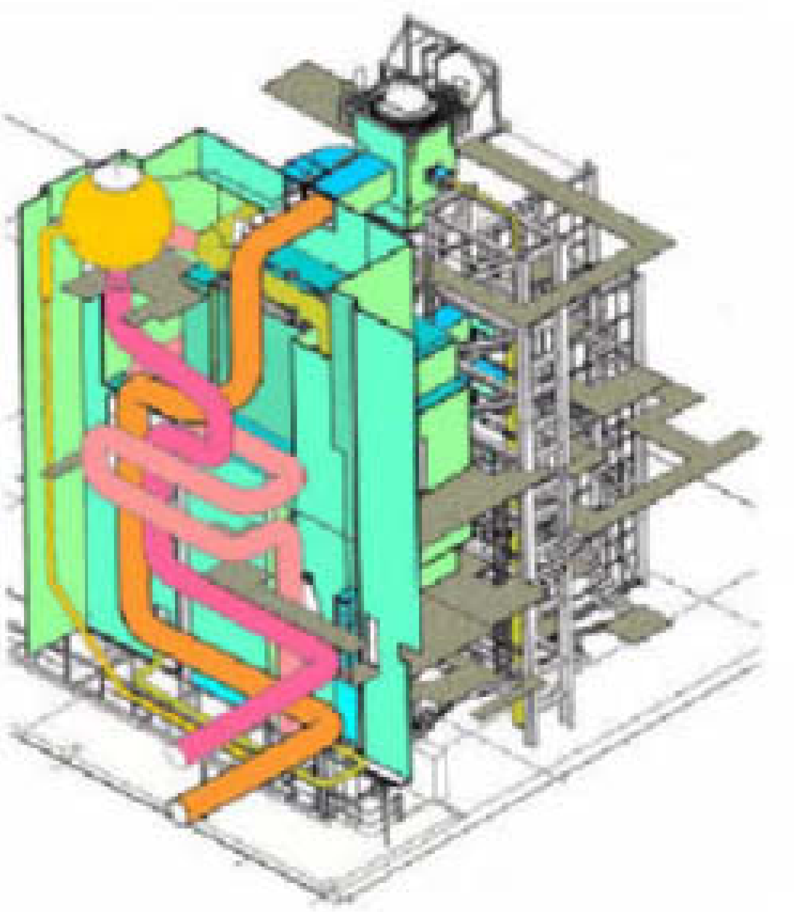


FIG. 1. Sodium fire protection in the steam generator building.

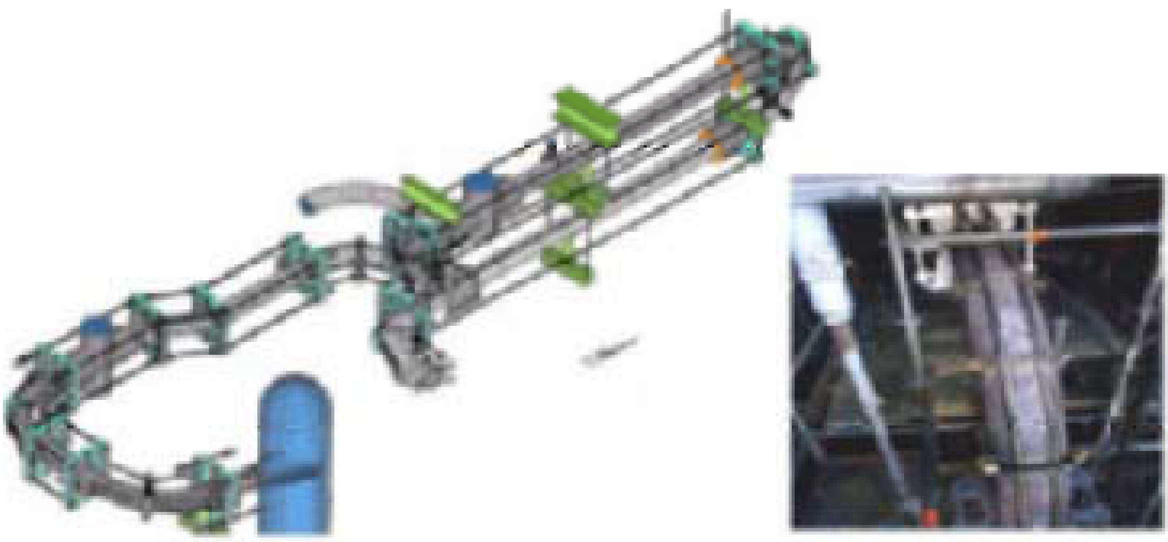


FIG. 2. Anti-whip system on the high-pressure steam pipes.

Reinforced earthquake resistance of the plant buildings. Given the various types of constructions on the site, dating from the 1970's, a specific approach was undertaken. There are pre-stressed concrete buildings, mixed concrete and steel frame building superstructures, pillar with independent base plate foundations, and mixed raft and pillars foundations. The earthquake behaviour of all the buildings as thoroughly analyzed, based on the original civil engineering plans and the expert evaluation of the building construction. A building reinforcement feasibility study was proposed to the Safety Authorities, based on the results of the earthquake calculations, the sensitivity analyses, feedback from earthquakes occurring on this type of construction and experimental results. The selected reinforcements, primarily involved the following anchoring of the pillars bases in the steel frames:

i. The stability piers and the steel ties;
ii. The concrete buttress walls;
iii. Roof ties;
iv. The concrete pillars reinforcing steel overlapping;
v. The separation of the buildings.

Specifically, the superstructure of the steam generator building had to be cut into two parts to separate it from the turbogenerator building (Fig. 3).

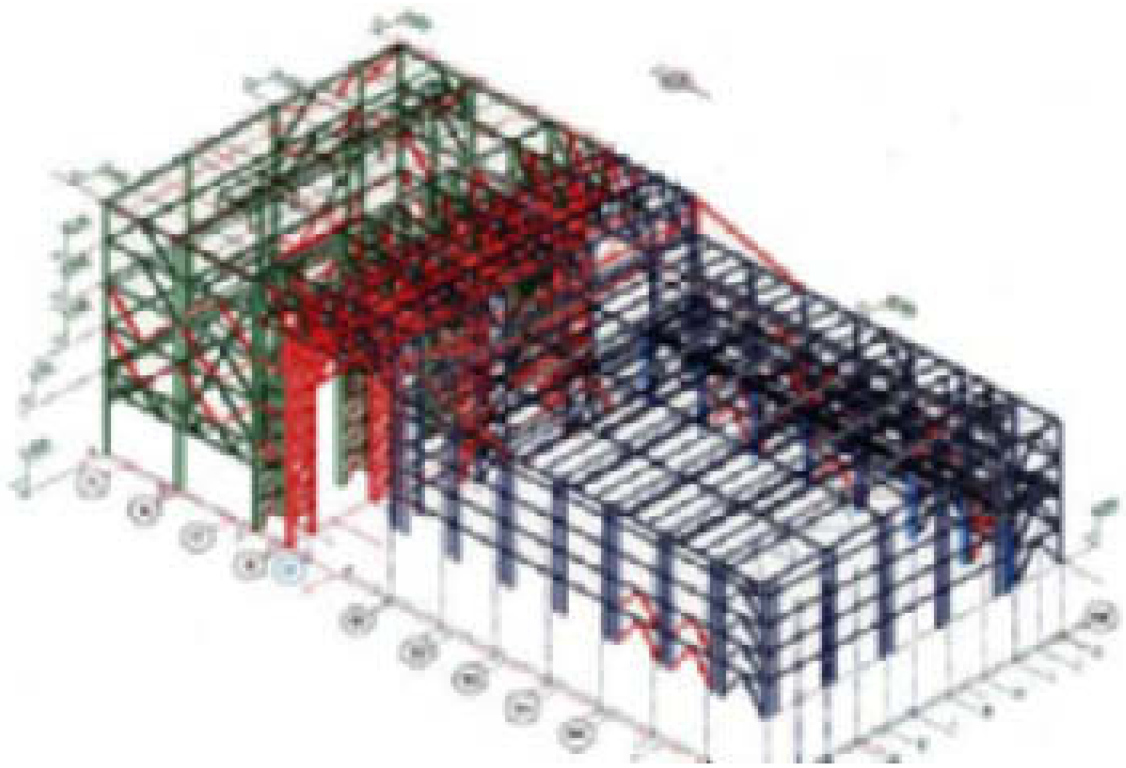


FIG. 3. Separation of the SG building from the turbogenerator building.

4. EXPERT EVALUATION AND FEEDBACK ON EQUIPMENT BEHAVIOUR

A major evaluation campaign was conducted on all the materials essential to safety. This campaign involved the secondary circuits, the steam generators, the primary sodium pumps, the intermediate heat exchangers, the sodium valves and fittings, and the reactor block internal structures. Systematic research was conducted into damage mechanisms related to thermal fatigue, creep fatigue, creep, inter-crystalline corrosion, high temperature intergranular cracking. This research focused on a wide range of materials present at Phénix: austenitic steels with low or average carbon content, titanium stabilized or not, chromium, nickel and molybdenum steels in the form of welds, non-alloy or low chromium and molybdenum-alloy ferritic steel, austenoferritic steels in the form of cast products.

The properties of the base materials and of the welded joints were tested in aged condition in terms of tension, cyclic behaviour, fatigue and creep characteristics. These materials were compared to materials in new condition in each field, in order to support dimensioning data and potential spreading of defects analyses.

Following this expert evaluation campaign, the defective equipments were repaired, replaced or left as is and justified by a non-propagation analysis for the defects, which backed up the non-destructive testing done on the site. In particular, the materials in 321 steel (Z6 CNT 18-10) displayed several defects of the relaxation cracking type, which led to systematic replacement of such secondary piping and the Steam Generator sodium headers.Likewise, the inspection of several Steam Generator modules resulted in the systematic repair, throughout all the modules on the superheater and reheater stages of the two Steam Generators Units, of certain zones in the hot part of the outside envelope of the modules which presented a risk of relaxation cracking (Fig. 4).

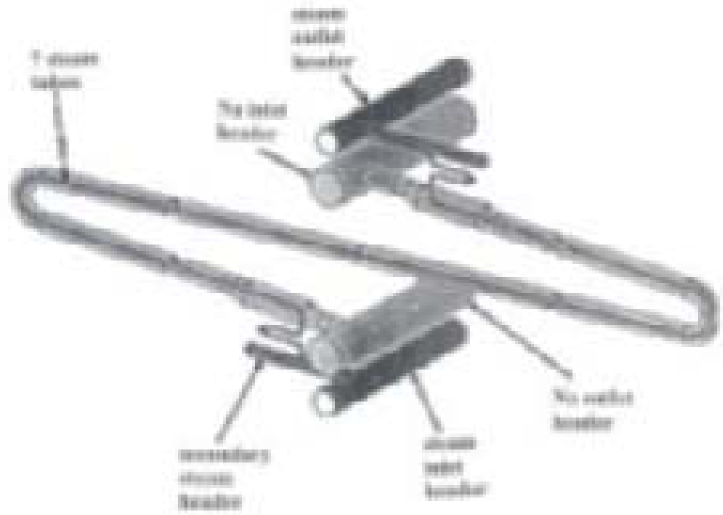


FIG. 4. Steam generator module.

This repair consisted of replacing the circumferential welds on the hot sub-headers and the hot bends. In addition the longitudinal welds on the hot subheaders and the cold bends were also examined. This repair, currently underway, involves the following operations:

- Disassembly of the modules;
- Washing the modules to remove residual sodium;
- Actual metallurgical repair of each module: cutout of the sodium envelope, assembly and welding of the new parts, and examination of the non-replaced parts mentioned above;
- Inspection of the new welds;
- Leak resistance test and hydraulic test for each repaired module;
- Reassembly in the Steam Generator.

5. THE REACTOR BLOCK STRUCTURES

The Phénix Reactor Block is the integrated type, designed in the 1960’s. In order to extend operations with respect to the current dimensioning rules, a special approach was implemented, taking into account the possibilities for inspection of the structures in-situ. This approach took into account the extensive feedback from operations that were provided by the analysis of the negative reactivity incidents that affected the reactor in 1989 and 1990. The approach was structured into three levels of analysis conducted on the main structures making up the reactor block. These levels are described in the following.

5.1. Re-evaluation of past and future damage to the structures

This required:

- Construction archive search for the manufacturing conditions in order to know: the properties of the materials, the chemical treatments used for cleaning, the heat treatments, the manufacturing methods, the inspections and repairs performed.
- Determination of the mechanical behaviour of some of the mechanical joints. Such was the case for the welded joints, which were of a different design than that recommended by the RCC MR construction code. These joints had to undergo additional evaluations in order to bring knowledge of their behaviour to levels thatcomply with current requirements. Major experimental programs integrating reactor environment conditions were conducted on the corner welded joints, with and without back of weld reworking, with respect to resistance to excessive deformation, to fatigue, and to creep fatigue.
- Determination of the loads through analysis of the reactor operating parameters combined with thermal hydraulics calculations for the specific thermal stratification and fluctuation areas. For this purpose, the recent R&D progress made in the EFR (European Fast Reactor) project was used and transposed to a real installation.

5.2 Verification of acceptable tolerance of the structures with respect to potential defects and the acceptability of anticipated failures

Given its importance with respect to the main safety functions, in particular reactivity control, the core support line, which includes the upper hangers, the main vessel, the conical shell and the diagrid, was the subject of in-depth design research. For the main vessel, the analyses conducted were based on a leak before break scenario, and on fracture mechanics for the other components. Given the high tolerance for conical shell defects, the fracture mechanics calculations focused on very large through faults, requiring mixed shell-bulk modelling for the entire reactor block in order to correctly represent the boundary conditions.

5.3. In situ monitoring of the large reactor block structures

In order to strengthen a first line of defence made up of the acceptability of structure damage during past and future operations and structure defect tolerance, an in-situ inspection programme was carried out on the major reactor block structures: upper hangers, conical shell and the core cover plug.

- The reactor block is suspended to the slab via 21 upper hangers. These hangers have three welds that are difficult to access, one of which is heterogeneous. A test programme for these three welds was developed on a model. Applying this to the reactor required the development of automatic ultrasonic inspection equipment;
- The conical shell supports the core diagrid on the main vessel of the reactor block. It has two full penetration welds at the connection points with the diagrid and the vessel. A third, non-penetrated weld provides the connection with the hydraulic baffle plate. Inspection of these hard to reach welds, which are several meters from the outside surface of the main vessel, required the development of a unique ultrasonic examination technique, using the sheet metal as the wave guide (Fig. 5). Carriers, able to cover 1/5th of the conical shell's circumference inside the 10-cm deep inter-space, support the ultrasonic sensor, operating at 150°C, in contact with the main vessel in the tip of the conical shell. These carriers are introduced into the vessel inter-space through 5 nozzles specially created for this inspection. This intervention is highly automated due to the location 10 meters under the reactor slab, inside the primary containment vessel, in a hot and irradiating environment. The preparation of this intervention required the development and validation of the ultrasonic examination techniques, cutting, and welding of the nozzles on the safety vessel on a full-scale model.
- The core cover plug located above the core ensures the guiding of the reactor reactivity control rod mechanisms and the positioning of the core assembly thermal instrumentation and of the burst pin location system. Visual inspection was conducted on this component, using high-resolution vision instruments operating at 150°C. This test first required partial drainage of the primary sodium from the reactor block, to the level of the sub-assembly heads (420 m^3 of sodium transferred). During this test, other structures in the upper part of the reactor block were examined: core subassembly lattice, primary vessel separating the hot and cold pool, fuel transfer machine (Fig. 6).

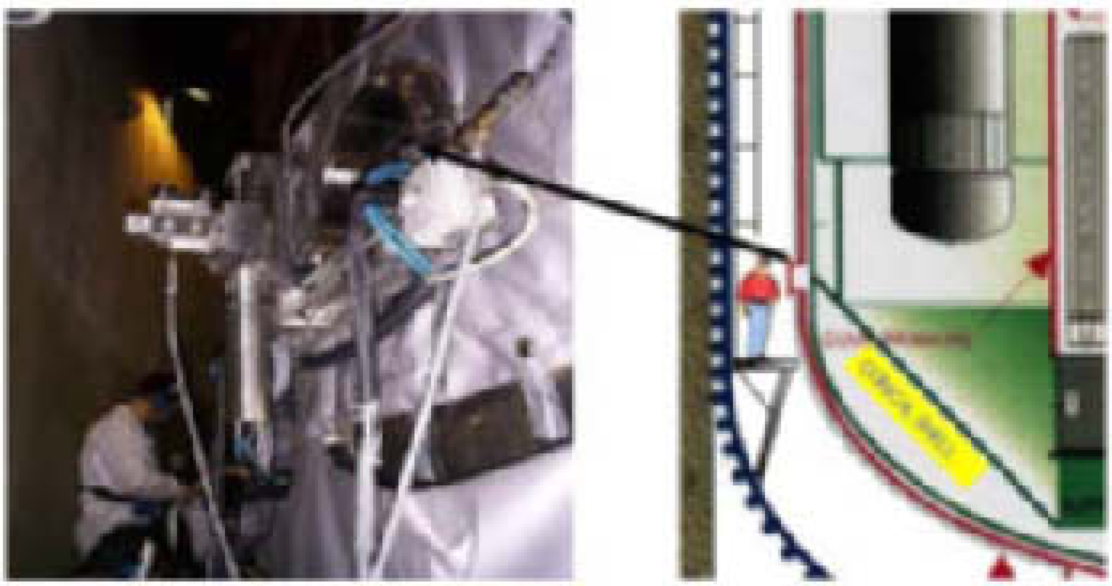


FIG. 5. Conical shell inspection.

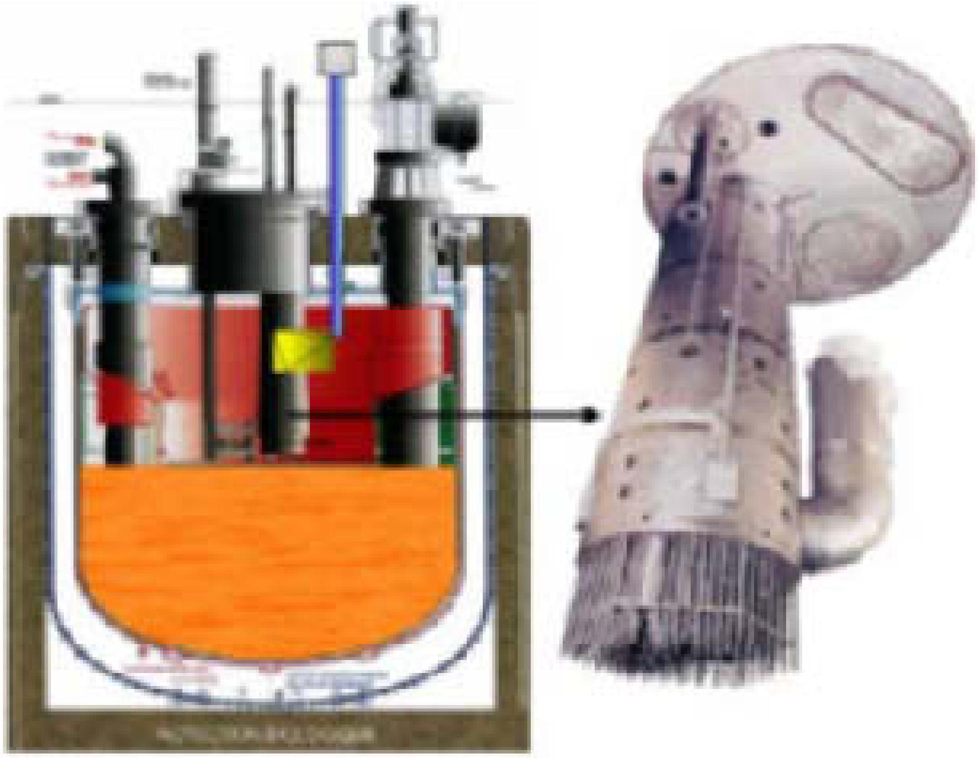


FIG. 6. Core cover plug inspection.

6. CONCLUSION

A vast modernization programme at the Phénix nuclear power plant has been undertaken. The works which have been performed or which are currently underway account for approximately 3 million hours. Power operation is scheduled to resume at the end of 2002.The reactor then will be able to carry out the irradiation experiments in order to provide answers on the subject of the future of nuclear waste. Furthermore, it will contribute to the acquisition of fast reactor operating experience.

In addition, important knowledge has been gained from the expert evaluations conducted on the materials and the components, after 100 000 hours of operation in real use conditions. Likewise, the methodology developed to extend plant lifetime, and the development of special inspection tools have resulted in significant progress for R&D and greatly increased expertise from which the entire nuclear programme will benefit.

ADVANCED AND INNOVATIVE APPROACHES TO INSPECT THE PHÉNIX FAST BREEDER REACTOR

M. GIRAUD, P. MAJOR
Framatome-ANP/Novatome, Lyon Cedex, France

J. GROS, L. MARTIN
CEA-Phénix, Bagnols-Sur-Cèze Cedex, France

PH. BENOIST
Commissariat a l'Energie Atomique (CEA), Saint-Paul-Lez Durance Cedex, France

O. BURAT
Framatome-ANP, Lyon Cedex, France

Abstract

Phénix is France's experimental fast breeder reactor that is located in Marcoule and is operated by the French Atomic Energy Commission (CEA). The CEA, has decided to renovate the aging plant to extend the life expectancy of the reactor. The program requires field inventory and inspection of equipment to support the component life-span evaluation studies. The presence of liquid sodium and the high temperature of the reactor coolant system even at cold shutdown - between 150 and 180°C - make inspection of Fast Breeder Reactors (FBR) a difficult technical challenge. Framatome-ANP has in particular completed four innovative non-destructive tests at Phénix:

- Inspection of the conical shell that helps to support the reactor core. The technique, which was developed in concert with the CEA/STA, was performed in late 1999. It involved ultrasonic testing of welds immersed in liquid sodium and located several meters from the scan surface.
- Inspection of the top portion and the hanging shells of the main vessel using small transducers able to withstand temperatures of 130°C.
- Inspection of the tubesheet in an intermediate heat exchanger using a photothermal camera.
- TV inspection of the primary circuit internals through dedicated periscopes, that needed the partial draining of the primary sodium, and was successfully achieved in April 2001.

1. INSPECTION OF THE CONICAL SHELL

This was performed as part of the inspection program of the reactor. It was designed to demonstrate that the core support structure was still in good health. The welds of interest, named 86, 85, and 83 (Fig. 1) are located 125, 300, and 3600 mm, respectively, from the outside of the main reactor vessel that holds the sodium. The welds are submerged in the sodium coolant.

After the initial test program, it was evident that only ultrasonic testing with a technique requiring contact with the main vessel could get the job done.

1.1. Inspection procedure

The inspection procedure was developed jointly by the CEA and Framatome-ANP. It consists in transmitting an ultrasonic signal, at different frequencies depending on the weld of interest, from the outer surface of the main vessel. The system then detects any echoes that would be reflected by a flaw in the main vessel.

The signal acquisition system used for the inspection was redesigned to accept the long delays for signals echoed from the far side of the structure (3600 mm away from the signal entry point). The processing software was developed from the CIVA program by the CEA/STA so that it processes ultrasonic signals after a brief delay.

Because there can be differences in the material permeability, the team studied the echoes produced by the geometry of the bottom flange of the structure. The study performed with the CIVA program demonstrated that there were no stray signals at any point around the edge of the vessel and that there were no through defects in any of the three welds inspected.

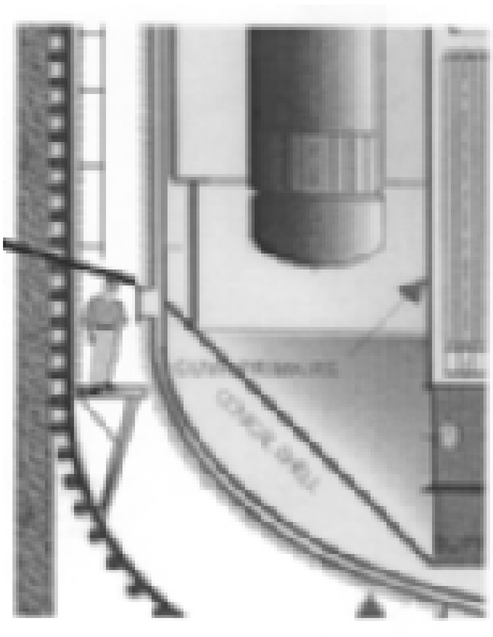


FIG. 1. Inspection of the conical shell — position of the inspected welds 83, 85, 86.

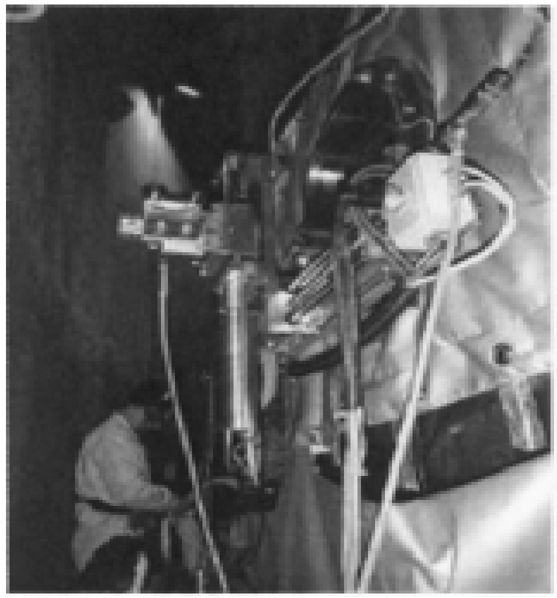


FIG. 2. Inspection of the conical shell — short carrier operating from the vault.

1.2. Testing

The Phénix plant operator asked Framatome/Novatome to make five penetrations through the double barrier around the main vessel so that the transducers could be placed against the outside wall of the main vessel. With the five penetrations, 98% of the vessel periphery is now accessible. Two remotely operated carriers were used to place the transducers over the welds around the circumference of the vessel. The short carrier permitted inspection of a 500-mm band on either side of the penetration axis (Fig. 2). The long carrier permitted inspection of a 500 to 4600 mm band around the penetration axis. An As Low as Reasonably Achievable (or ALARA) approach was used. The design group was composed of the Phénix owner, including the plant health physics unit, the prime contractor Framatome/Novatome and the other principal contractors involved. They managed to cut the integrated dose to inspection personnel by a factor of 4 (during both the preparation and inspection phases).

1.3. Results

The set of test performed demonstrated that the inspection process had the desired level of performance. It can detect 100 mm long through defects in welds 86 and 85, and 200 mm long through defects in weld 83. All 97 m of weld were inspected in eight weeks. The transducers travelled 2.25 km over the wall of the 150°C main vessel. No indication of the type investigated was detected in the conical shell and the welds inspected. Review of testing records shows a good propagation of the ultrasonic waves in the inspected structure without any stray signals.

2. INSPECTION OF WELDS IN THE MAIN VESSEL HANGERS

The main reactor vessel is hung under the reactor slab by 21 hangers.

The inspection was performed to demonstrate that the welds in the hangers were free of defects. The hangers are made of 316 l steel. They are 60 mm thick and are at 110°C when the reactor is shut down. The structure is difficult to access and has only a 10-mm space on either side of the scanning surface.

In order to overcome these difficulties, the Framatome Technical Center developed a special 8-mm thick transducer that operates at temperatures between 110 and 130°C. It operates continuously and has a built-in feeding device for the coupling fluid (Figs 3 and 4).

The transducer qualification tests demonstrated that the sensors were highly sensitive, that they could withstand sharp temperature changes and that there was little change in sensitivity between 20 and 110°C. The sensors were used successfully to demonstrate that there were no cracks in the hangers. The inspection was performed to detect flaws at 1/20th of the thickness probed. The inspection required development of automated inspection facilities adapted to the characteristics of the different examined zones.

FIG. 3. Miniaturized transducerat 45.2 MHz.

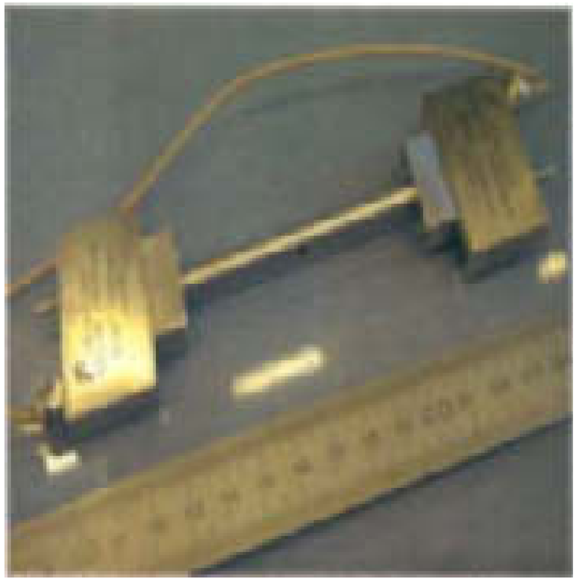


FIG. 4. Miniaturized double transducer OT 45.2 MHz.

3. INSPECTION OF AN INTERMEDIATE HEAT EXCHANGER TUBESHEET

3.1. Objective

The inspection was designed to demonstrate the absence of flaws in the tube-to tubesheet welds in the intermediate heat exchanger (IHX) at the Phénix fast breeder reactor. Initially, the Framatome Technical Center needed to industrialize the inspection system, which used an Active Photothermal Camera (APC). The inspection replaces dye penetrant inspection to detect emerging defects and it can detect defects beneath the inspected surface if the top ligament is about 0.5 mm thick. This type of defect (inclusion) is usually detected by radiographic methods, but radiography of the intermediate heat exchanger would have been time consuming and would have exposed personnel to high doses of radiation.

3.2. Basic principle

The Active Photothermal Camera (APC) analyses propagation of heat produced by a laser aimed at the surface of the part being inspected. The analysis is performed by an infrared detection system that can pinpoint a disturbance in the heat transfer field such as that caused by a defect, whether emerging or subsurface.

The laser excitation beam and infrared detection beams are moved together and sweep the inspection surface in a square sawtooth pattern. The image of the part built up from the scan shows any flaws in the part.

The photothermal camera can be used in addition to, or in lieu of, traditional surface inspection techniques (dye penetrant, magnetic particle or eddy current examination). The principal characteristics of the "Standardized Flying-spot" technique used in the inspection process enable the photothermal camera to:

- Identify the following without touching the part (it remains between 10 and 60 cm from the surface)
    i. Flaws with openings of several micrometers;
    ii. Flaws under a surface ligament as well as emerging defects;
    iii. Flaws on rough surface (oxidation, machining, surface roughness Ra < 12).

- Be able to inspect the following irrespective of their magnetic properties:
    i. Insulation material (dielectrics);
    ii. Hot components.

The procedure does not, therefore, contaminate the surfaces inspected and does not remove material from the surface or disturb it. This means it does not contaminate work areas.

3.3. Industrialization

Industrialization of the equipment consisted in miniaturizing the photothermal camera so that the entire inspection head could be mounted on a remote-controlled XV manipulator (meaning that the operator does not need to perform any work close to the controls).

The manipulator is precise enough to position the photothermal camera over each 100×100 mm zone of the tubesheet. The photothermal image of the entire zone is generated directly by the inspection head.

3.4. Performance obtained for IHX inspection

3.4.1. In the laboratory

- Detection of flaws under 0.5 mm-thick ligaments.

3.4.2. In the field

- Inspection of two 1.2-m diameter tubesheets (Fig. 5);
- 2300 welds inspected per tubesheet (Fig. 6);
- Weld acquisition rate: one weld/minute.

FIG. 5. Photothermal camera operating on site.

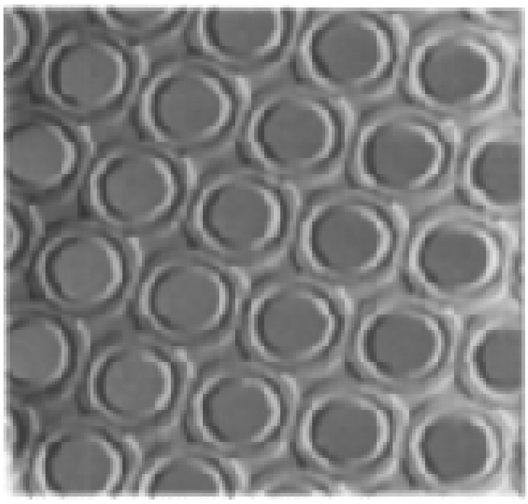


FIG. 6. Photothermal image from an inspection sequence.

4. THE ONE-OFF TV INSPECTION OF PRIMARY CIRCUITS INTERNALS

After three years' preparation, the TV inspection of the primary circuits internals has been successfully issued in April 2001. The Novatome division of Framatome-ANP, completed the extensive non-destructive inspection program requested by the Phénix plant with an inspection of the primary circuit internals. Included in its service package were the setting up of the TV inspection — which required the partial draining of the primary-side sodium — the supervision of the supply of various materials needed to be developed and qualified, and the actual carrying out of the inspection. Justification studies for the inspection were carried out at the same time focusing on the thermalhydraulic, safety and radiation protection aspects affected by the lowering of the sodium level in the primary vessel.

4.1. A complex inspection

The inspection involved draining the 420 m^3 sodium in the reactor until it was levelled with the tops of the subassemblies, carrying out the visual inspection of the reactor internals, the sub-assemblies network, the core cover plug and the core instrumentation support grid using TV, and then raising the sodium level in the reactor to its normal level. The temperatures reached by the sodium and the cover gas of argon (180 and 140°C), and the increase in radiation due to reducing the level of sodium, mean that high-performance equipment must be used, in particular shielded, vacuum-shrouded periscope, so that the video equipment can be kept outside the reactor. As well as being technically complex, the inspection also involved managing numerous interfaces with the inspection teams, reactor internals and inspection equipment. The organization set by Framatome ANP for the inspection itself required mobilizing a team of 28 qualified personnel in 3×8 hours shifts, 7 days a week over 26 days, with each member being given three weeks' training beforehand.

4.2. Satisfactory results

The inspection met the planned deadlines and its scope was extended beyond the initial objectives focussed on the status of the core instrumentation (thermocouples and the support grid) and the above core structure due to the quality of the image obtained. The results of the inspection confirmed that the internals were in good conditions.

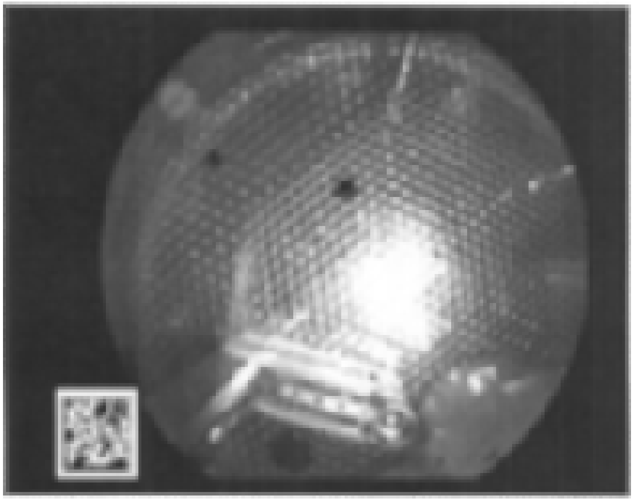


FIG. 7. Inspection of the subassemblies network.

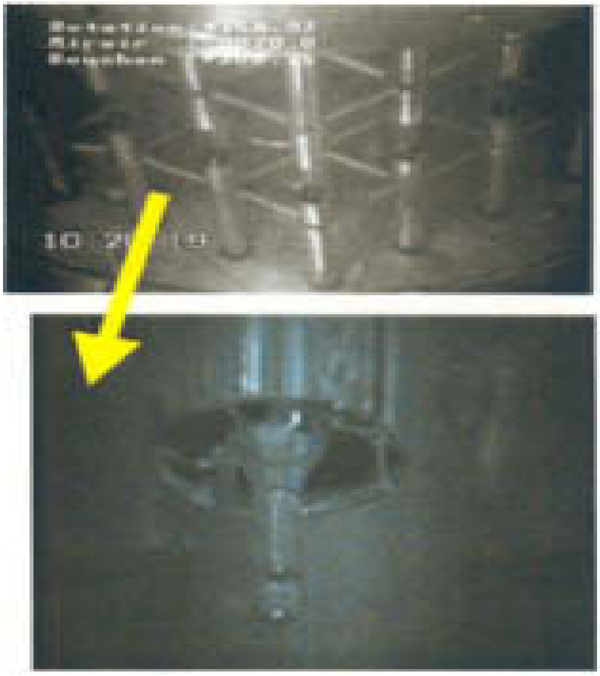


FIG. 8. Inspection of the core instrumentation.

FIG. 9. TV periscope for high resolution images.

5. CONCLUSION

The objectives of the different inspection operations discussed here were achieved. The solutions involved development of innovative technical solutions to withstand the special operating conditions imposed by work in a Fast Breeder Reactor, i.e. the presence of liquid sodium, the high temperatures of the components, even during shutdown and the stringent health physics limitations. The processes demonstrated a high level of operability during the different inspections and the results obtained attained the requisite levels of quality. A major step has thus been taken, bringing the startup date for Phénix closer. Moreover, it leads to a great amount of experience in the field of In Service Inspection of Fast Breeder Reactors.

SODIUM CLEANING IN PHÉNIX STEAM GENERATOR MODULES

O. GASTALDI
Commissariat a l'Energie Atomique (CEA), Saint-Paul-Lez Durance Cedex, France

V. GRABON
Framatome-ANP, Lyon Cedex, France

CH. CAVAGNA
CEA-Phénix, Bagnols-Sur-Cèze Cedex, France

Abstract

The repair of the Phénix steam-generator's modules needed to establish a particular sodium cleaning procedure. The standard Phénix cleaning procedure, using water vapour in presence of nitrogen and carbon dioxide was indeed not suitable, because of bulky sodium residues and specific geometry of modules. A specific procedure, using a gradual water vapour nitrogen process has been developed, qualified and industrially implemented in a very short delay, in order to clean efficiently the steam generators modules, before repair and requalification.

1. INTRODUCTION

With regard to the life-time extension of the FBR Phénix, metallurgical examinations have been carried out on the Steam Generator (SG) in 1998-2000. Some cracks have been detected in the hottest parts of the SG. It has been decided to repair all superheater and resuperheater of the SG.

Because of the high reactivity of sodium with air and caustic corrosion risks that may arise during repair and further operations, due to sodium/moist air reaction products, residual sodium had to be eliminated in order to achieve the requalification of the SG before start up. For this, a particular sodium cleaning procedure has been established, qualified and industrially implemented in a very short delay in accordance with Phénix start up planning.

2. SELECTION OF THE APPLIED SODIUM CLEANING PROCESS

The various ways to eliminate sodium residues are:

Physical ways:

- Blast cleaning;
- Scraping: only for local use on accessible and little quantities of sodium;
- Evaporation under vacuum: only on small material.

Chemical ways:

- Reaction with water: usual methods in 90% of cases;
- Reaction with alcohol: no more allowed in France;
- Reaction with other products: "exotic" methods.

2.1. Sodium cleaning background

In France methods using water/CO_2 mixture are the usual ways of cleaning. For example Phénix and Superphénix components (pumps, exchanger), fuel assemblies are cleaned in specific pits with water vapour or sprayed water in presence of CO_2. A particular procedure of carbonation has been achieved in situ on the "Barillet" of Superphénix in 1989 before dismantling of this component. Up to now no cleaning operation had been performed on SG in France with the purpose of the component's requalification

In other countries Water Vapour Nitrogen type processes (WVN) are generally preferred for sodium removal. Few data about some SG cleaning experiences have been reported. Generally the background about cleaning with water processes is good. Meticulous rinsing is recommended following cleaning. Water is a common, cheap product. The products of the sodium/water reaction: sodium hydroxide and hydrogen are easily managed but the reaction must be controlled. The use of CO_2 lowers the caustic corrosion risk by conversion of the sodium hydroxide into solid and chemically inactive products: sodium carbonates. However, this product may form a barrier between the sodium and the reactant gas likely to slow the process kinetic. Carbonation processes should be adapted case by case.

2.2. The Phénix Steam Generator (SG) particular geometry

The reactor is cooled by 3 secondary sodium circuits each of them having one SG of modular concept. Each SG is composed of the economizer-evaporator, the superheaters and the resuperheaters stages; each of them is composed of 12 modules. Forty-seven modules from 2 SG's superheater and resuperheater have to be repaired. A superheater module has the S shape and is composed of a 27 m long-ϕ 200 mm shell, containing 7 water vapour tubes (Fig. 1). The sodium flows around the tubes. The tubes are maintained by regularly spaced grids.

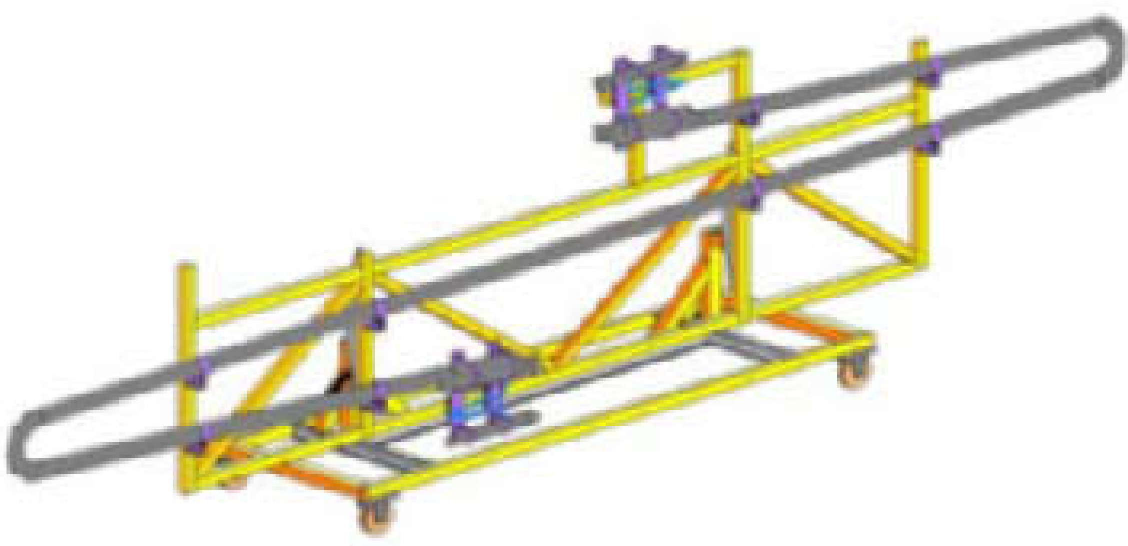


FIG. 1. Module on transfer system.

After draining of the SG, only a thin film of sodium lays on the tubes external walls and on the shell internal wall. In the lower part of the superheater modules bulky sodium residues (until 5 cm thickness-65 g/sleeve) may be present in the 7 thermal protective sleeves around the tubes (Fig. 2). These sleeves are quasi closed spaces with a reduced annular gap less than 2 mm large.

The total sodium residual mass may vary from some hundred of grams to several kilograms depending on the draining conditions and on each module geometric particularities such as misalignment.

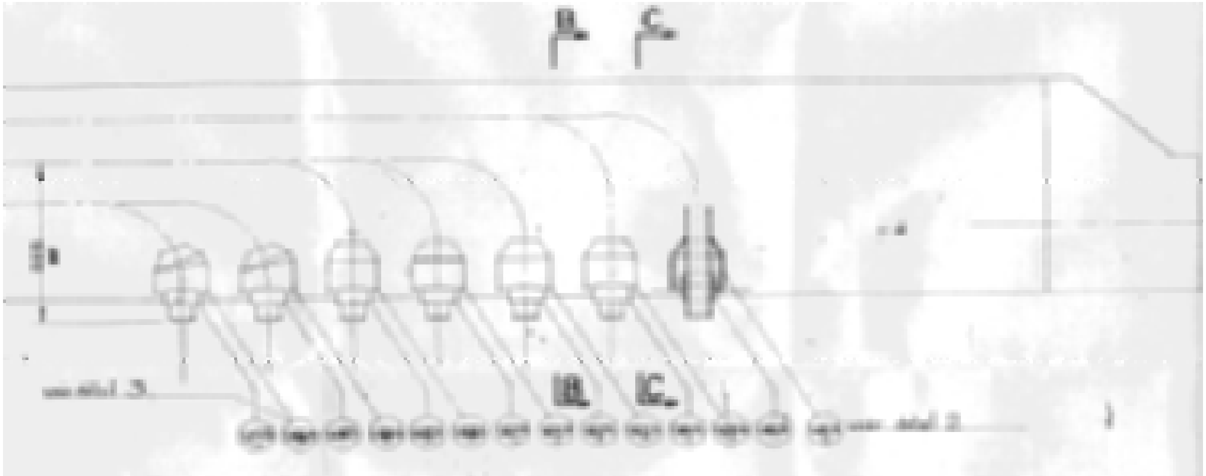


FIG. 2. Detail of the sleeves area.

2.3. Definition of the SG cleaning way

Local cleaning of the zones to be repaired with isolation of non-repaired area under inert atmosphere was not an acceptable way because it required longer development to achieve the tightness in such a complex geometry. Then it was decided to clean the entire module with water process. Because of the presence of bulky sodium residues in the sleeves, a procedure using nitrogen as water transport gas was preferred rather than CO_2, in order to produce liquid reaction product (aqueous sodium hydroxide) that may flow outside the sleeves and allow a complete elimination of sodium in a reasonable delay. The process has been adapted to avoid the risk due to aqueous product flowing on metallic sodium. Further draining of sodium or scraping was not possible because of the too narrow access gap.
The sodium/water reaction characteristics are:

- Immediate and exothermic reaction (138 kJ/mol);
- Production of an explosive gas (hydrogen lower inflammable limit is 4% in air);
- Production of a corrosive product (aqueous sodium hydroxide). Caustic stress corrosion cracking risks arise at around 120°C on stainless steel (Phénix superheater and resuperheater material is an austenitic stainless steel).

The chosen process has to deal with all these difficulties:

- The global kinetic of the sodium elimination rate can be controlled by the reactant (water) supply: water is introduced in the form of water vapour with gradual increasing concentration.
- The provisions made against hydrogen risk consist to clean under inert atmosphere (N_2), to limit the H_2 maximal theoretical concentration to 1% in the effluent gas by limiting the nominal water concentration in the first steps of the process and to measure continuously the H_2 concentration in the effluent. Alarms and automatic actions are triggered if the H_2

level reaches 3%. Otherwise, to prevent air entrance during the treatment, the module is always kept in overpressure with regard to atmospheric pressure.

- The caustic corrosion risks must be limited at every step:
  i. Before cleaning: the module is maintained under inert atmosphere to avoid the formation of aqueous sodium hydroxide due to air ingress.
  ii. During cleaning: as aqueous sodium hydroxide is present, the preventive measure against corrosion risk consists to control the sodium/water reaction exothermicity in order to maintain the temperature under 60°C. This is achieved through the control of water supply into the module.
  iii. After cleaning: sodium hydroxide residues must be eliminated as far as possible by meticulous rinsing to limit caustic corrosion risks during further heating (welding during repair, heating of SG before start up). This is achieved by filling the module with liquid water and circulation of water during long cycles (total time 100 hours). This procedure may allow NaOH residues to diffuse from any eventual defect and to be mechanically removed by water flow.

Additional provisions should be made to avoid hazardous violent reaction- sodium/liquid water type- due to uncontrolled flow of aqueous sodium hydroxide on metallic sodium retentions. Bulky sodium residues should not be on the flowing line of liquid effluent. That is why the module could not be treated in the SG in-situ vertical position. An inclination of the module of 30° with regard to the vertical position is needed. For the same reason, the reactant gas should flow from the lower inlet to the upper end of the module, because the water vapour reacts immediately with the first encountered sodium surfaces as it enters into the module.

2.4. Validation tests

In parallel with the cleaning procedure definition, validation tests have been carried out at CEA Cadarache in order to confirm the chosen options and to optimise the process parameters;

- Draining tests of sleeve filled with liquid sodium were not successful and confirmed that further draining of sodium is hazardous.
- Cleaning tests on thick sodium samples to confirm the kinetics of sodium elimination.
- Cleaning efficiency tests on Phénix module singularities (sleeve, grids, flow mixing device). The test showed that a global flowing of the reactant gas in nominal conditions has few efficiency on the elimination of the sodium inside the sleeves. A local treatment is necessary to achieve the cleaning in a reasonable delay (about 30 hours) and without unacceptable temperature increase. This is performed by a local injection of the reactant gas inside the sleeve. For better access and efficiency a hole has been machined in the sleeve and the injection system through this hole has been optimised.

2.5. SG cleaning procedure

General requirements:

- The module is initially under inert atmosphere;
- Holes have been machined in the 7 sleeves;
- The module is inclined at 30° with regard to the vertical position;
- The cleaning phases have to be continuously carried out;
- Demineralized water is needed.

The cleaning procedure has been defined as follows:

- Global cleaning with wet N_2 sweeping: to treat the thin residues of metallic sodium (module's position: 30°);
- Local cleaning with wet N_2 sweeping. The 7 sleeves are treated in parallel with a specific rod: to treat the bulky sodium retentions (module's position: 30°);
- Local cleaning with sprayed water in N_2 flow: to complete the treatment of the bulky sodium retentions (module's position: 30°);
- Global cleaning with sprayed water in N_2 flow: to complete the treatment of metallic sodium (module's position: 30°);
- Total immersion: to demonstrate the absence of metallic sodium (module's position: 30°).

The rinsing of sodium hydroxide is performed by:

- The previous total immersion of the module with water and water circulation, the module being inclined at 30°;
- Local aspersion of sleeves with liquid water injection (module's position 30°);
- Total immersion of the module with water and water circulation. The module is in horizontal position to allow complete immersion of the sleeves.

After draining water, the module is dried prior to be repaired.

3. INDUSTRIAL FACILITY

3.1. Intellectual pathway

Before to apply the treatment on all the modules to be repaired, experimental operations have been done at an industrial scale on 2 modules to qualify the cleaning procedure. The different stages of the procedure have been tested and the process has been assessed through the different measurements (residual sodium mass, hydrogen release, temperature increase, efficiency). To achieve this first action, a prototypic cleaning unit has been adapted on a cleaning pit of PHÉNIX. This facility has been developed to offer a great flexibility of use and to ensure a very well controlled process. This facility allows injecting the different gas mixtures and water in the module. It is the same facility that allows to clean and to rinse the module. Once the process well qualified with both these tests modules, the definition and the building of the final industrial facility can be launched.

3.2. Process qualification

For these qualification the chosen process is composed of 10 phases:

1. Sweeping with an inert gas;
2. Tightness test;
3. Global sweeping with wet N_2 (low dew point) in the module;
4. Sweeping with wet gas with local injection in the sleeves;
5. Local Sweeping with sprayed water/N_2 in the sleeves;
6. Global Sweeping with sprayed water/N_2 in the whole module;
7. Slow filling of the module and water forced convection;
8. Local water injection in the sleeves;
9. Total filling with water and water forced convection (five cycles);
10. Drying.

The total duration of the cleaning phase is about 60 hours and the cumulative duration of the rinsing phases is about 100 hours.

3.2.1. Main measurements and instrumentation

The main parameters to consider during the cleaning of the module are:

- The temperatures of the module in different points corresponding to the potential sodium retention where more reactivity is expected;
- The hydrogen concentration in the gaseous effluent, which shows the progress of the reaction;
- The oxygen fraction which allows to assess inert atmosphere condition to avoid potential inflammation of hydrogen;
- The pressure in the module which has to be close to 30 mbar to ensure that there is no air entrance.

During the rinsing phases the main parameters are:

- The conductivity;
- The pH;
- The water temperature.

To pass from one phase to another one, several criteria have been fixed. A minimal duration is required for each phase. In addition, at the end of each cleaning phase the hydrogen concentration must be under a specific threshold, and no hot area must be detected. During the rinsing phases water quality criteria are required at the end of the 2 last phases.

3.2.2. Results and qualification

During both the cleaning operations the hydrogen signal never exceeded 0.2mol.%. Temperature increase is very limited. It shows that the reaction between water and metallic sodium is very slow, and that there is no uncontrolled reaction. The total amount of removed sodium which represents the total amount of residual sodium in the modules of interest was evaluated to about 400 g.

From one rinsing phase to another one the conductivity measured in the water effluent decreases quickly. In the fifth rinsing phase, the water conductivity is around 1 µS/cm, which shows that the sodium hydroxide removal becomes insignificant. After drying, no sodium hydroxide has been detected. The different assessments show that the sodium hydroxide removal is total. Then the development of and industrial facility was undertaken.

3.3. Industrial facility

3.3.1. Description

In order to limit the total duration of cleaning operations, the modules are treated in parallel on 2 lines. The facility is composed of two cleaning lines, two rinsing lines and a drying area as drying duration is relatively short. By this way, all the modules can be treated according to the start-up planning of Phénix. The two cleaning lines are set in a closed area equipped with heating system. It allows to work without condensation risk with gaseous water having a water fraction in the gas mixture about 2 mol.%. A global view of the facility is given in Fig. 3.

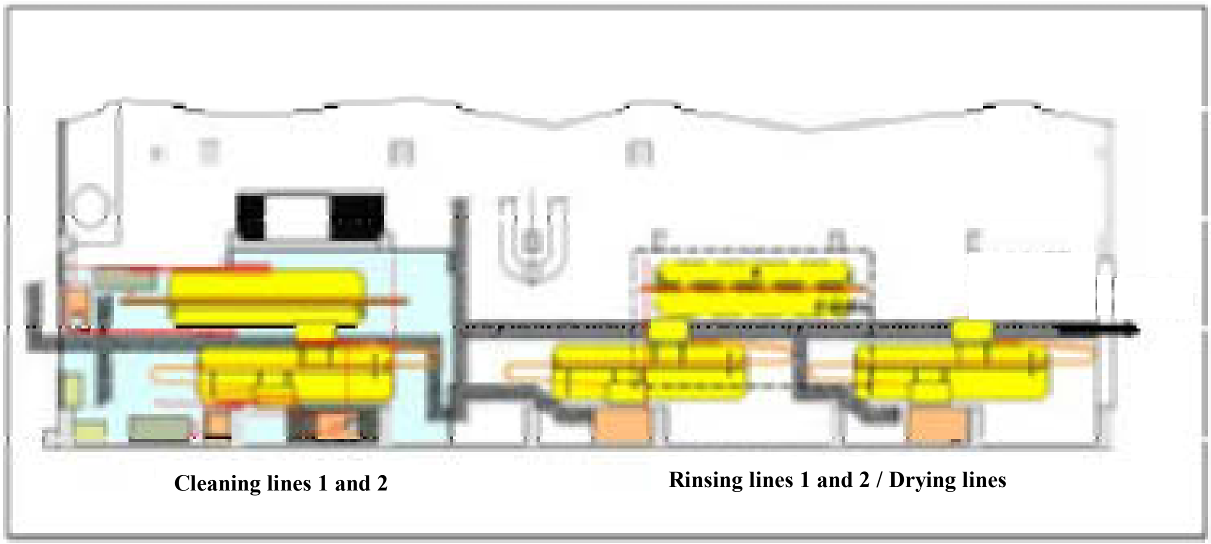


FIG. 3. Industrial facility.

Each cleaning line is composed of:

- A preparation system: to produce the specific gaseous mixture for each phase;
- A feeding system to connect the module;
- A specific tank to separate liquid and gas;
- An effluent line for the gas;
- An effluent line for the liquids.

The more complex part is the preparation system that gathers the different flow meters and the different control instruments.

3.3.2. First experiments feedback

More than 12 modules have been totally treated up to now. The hydrogen production has always been very low and controlled. The average duration of the cleaning is always quite the same (around 60 hours). The initial residual sodium quantity has never exceeded 400 grams. The temperature of the module always stayed under the criterion 60°C. The maximum temperature increase was about 20°C (for an initial temperature of 25°C).

4. CONCLUSION

The chosen process is well adapted to the specific geometry of the SG's modules. The low water fraction in the gas mixture allows to control in a good way the sodium water reaction without significant hydrogen concentration and temperature increase. The temperature is sufficiently low to insure there is no problem of caustic corrosion due to the sodium hydroxide during the cleaning phases. A long duration rinsing insures the total elimination of sodium hydroxide.

Up to now, all cleanings and all assessments are positive and a significant number of SG's modules have been already treated.

REVIEW OF FAST REACTOR OPERATIONAL EXPERIENCE GAINED IN THE RUSSIAN FEDERATION. APPROACHES TO THE CO-ORDINATED RESEARCH PROJECT

V.M. POPLAVSKY, YU.M. ASHURKO, YU.E. BAGDASSAROV, A.V. KARPOV, M.P. NIKULIN, A.M. TSIBOULIA
State Scientific Center of the Russian Federation (SSC RF) — Institute for Physics and Power Engineering (IPPE), Obninsk, Russian Federation

N.N. OSHKANOV
Beloyarskaya Nuclear Power Plant (BNPP), Zarechny, Russian Federation

A.I. KIRYUSHIN, B.A. VASILYEV
Experimental Design Bureau of Machine Building (OKBM), Nizhny Novgorod, Russian Federation

K.L. SUKNEV
Saint Petersburg Research and Design Institute ATOMENERGOPROEKT (SPAEP), Saint Petersburg, Russian Federation

A.S. KOROL'KOV
State Scientific Center of the Russian Federation (SSC RF) —– Research Institute of Atomic Reactors (RIAR), Dimitrovgrad, Russian Federation

V.V. DENISOV
Experimental and Design Organization "GIDROPRESS", Podolsk, Russian Federation

Abstract

The review of the experience gained in Russia in the field of fast reactors with sodium coolant is given in the report. The information on fast reactors operating in the Russian Federation (BR-10, BOR-60, BN-600) is presented: their current status, further prospects, and basic indices achieved by the facilities. The principal results of operation of test facilities and power plants with fast reactors in Russia are summarized. Necessity in implementation of special work on preservation and generalization of experience gained in the field of fast reactors have been analyzed, as well as possibility of organizing a coordinated research project in this area. In particular, possible approaches to the organization of activities on systematization of published information on fast reactors.

1. INTRODUCTION

In the early stage of development of reactor technology based on the use of fast spectrum neutrons, large efforts were made on choosing coolant that would meet fast reactor requirements to most extent. Among coolant candidates, liquid metals were considered (mercury, alkali metals and their alloys and heavy liquid metals), as well as gases (helium and carbon dioxide). Later, possibility of use of dissociating gases (nitrogen tetraoxide) as fast reactor coolant was studied. As a result of these studies, sodium was chosen as the most promising coolant for NPP with fast reactors from the standpoint of its neutronics, physical, chemical, thermohydraulic and economical parameters. Further, sodium was widely used as a coolant of stationary fast neutron reactors.

Fast reactor development during 50 years (about 350 reactor-years of operation) has been in general successful. Feasibility of fast reactor application for commercial electricity production has been proved. Reliable, stable and in general safe operation of fast reactors has been and operation has been based on the use of sodium coolant. Economical parameters of fast reactors are worse than those of thermal reactors, but this difference would decrease with expansion of fast reactor share in the nuclear power and turning from demonstration to commercial plants. Actually, nobody can predict now what coolant will be chosen in the future for fast reactors. It is quite probable that more than one coolant will be used in fast reactors in the future, however, in any case sodium cooled fast reactors (SFR) are expected to be claimed. In view of current pause in fast reactor development, the problem arises on preservation of experience gained in this area. The task of experience preservation should cover general aspects of fast reactor development, but not only their operation and decommissioning. The problem of experience preservation is urgent for each country involved in the development of this technology. In our opinion, international cooperation in this area will facilitate both exchanging gained experience and working out common approach to its preservation.

2. REVIEW OF EXPERIENCE GAINED IN FAST REACTORS IN RUSSIA

Comprehensive studies on fast reactors (FR) were initiated in the USSR in the 50-ies of the 20^{th} century. As a result of these studies, several FR facilities were constructed, beginning with experimental reactors (BR-5/BR-10 and BOR-60) to commercial reactors (BN-350 and BN-600). NPP with the BN-800 reactor is now under construction on Beloyarskaya site. Design studies are carried out on large size reactor BN-1800.

2.1. Experimental reactor BR-5/BR-10

2.1.1. The main characteristics of BR-5/BR-10 reactor

In January 1959, BR-5 reactor having 5 MW rated power was put into operation at the IPPE. There are three heat removal circuits in the BR-10 facility (sodium in the primary circuit, originally sodium-potassium and then sodium — in the secondary circuit, and air in the third circuit) with two parallel loops. Initial parameters of the primary and secondary coolants were respectively 430/500°C and 380/450°C, i.e. close to those of power FR. Now sodium temperatures in the primary and secondary circuits are respectively equal to 330/450°C and 270/370°C. There is a wide range of experimental devices in the reactor, namely test channels and irradiation devices and beams of thermal and fast neutrons. There are 5 dry instrumented channels in the reactor. Fast neutron flux in the central loop channel is up to 8.4×10^{14} n/cm^2·s.

2.1.2. The main stages of BR-5/BR-10 operation

There have been 4 main stages of BR-5/BR-10 reactor operation:

- 1959-1964: plutonium dioxide core loading, N_R = 5 MW;
- 1964-1971: uranium monocarbide core, N_R = 5 MW;
- 1973-1979: plutonium dioxide core, N_R = 8 MW;
- from 1983 till now – uranium mononitride core, N_R = 8 MW.

Fissile element enrichment is 90%. Updating of the reactor took place from 1971 to 1973 and from 1979 to 1983. The first redesign completed in 1973 envisaged increase of the reactor power up to 10 MW, and so it was titled BR-10. However, because of some problems related to reactor vessel air cooling system reactor power was limited to 8 MW level.
The following procedures were performed during the first redesign:

- Replacement of centrifugal pumps with electromagnetic pumps;
- Replacement of the primary coolant;
- Replacement of sodium-potassium alloy in the secondary circuit by sodium;
- Replacement of carbide fuel UC by plutonium dioxide;
- Installation of additional two sodium-air heat exchangers in the secondary loops.

Before this redesign high carbon content was observed in sodium (40–170 ppm) because of its high content in the initial sodium, and ingress of lubricant from the lower bearings of the centrifugal pumps.The above measures resulted in the decrease of carbon content in sodium down to 5-35 ppm.

In the course of the second overhaul of the reactor, the following main procedures were performed:

- Replacement of reactor vessel with compensating cylinder because of high neutron fluence;
- Turning from plutonium dioxide to uranium mononitride UN;
- Installation of safety jackets on all main and auxiliary pipelines of the primary circuit from reactor vessel to shut-off valves;
- Replacement of lyre-shaped sections (adopting thermal expansion) of the main primary pipelines by bellows;
- Increase of reliability of normal and emergency power supply systems of the primary pumps and other measures increasing reactor safety.

As a result of reactor modification, unique experience has been gained in dismantling and replacement of such large size component as reactor vessel.

2.1.3. The main results of operating experience of BR-5/BR-10 reactor

The principal objective of the BR-5/BR-10 reactor was to gain practical experience of operation of sodium cooled fast reactor and find optimum solutions of engineering and technological problems arising in SFR operation, which can be used in the future power reactor designs.

In order to solve these problems, the following wide range of studies have been and still are carried out on the reactor:

- Tests of various fuel compositions (plutonium dioxide, uranium monocarbide and mononitride) in order to study burn-up effect on the fuel characteristics;
- Study of various structural materials behavior as a function of fluence in fast neutron flux;
- Studies of corrosion effect of sodium coolant on structural materials;
- Mastering sodium technology including special aspects related to radioactive sodium;
- Mastering reactor technology and optimization of various systems and equipment;
- Operation with failed fuel elements (including tests of fuel elements with deliberately perforated cladding);
- Using reactor as neutron source (in order to produce various isotopes for medical and technological purposes, to produce nuclear membranes and to treat oncological patients).

A lot of information has been accumulated on tested fuel. Results of tests of advanced nitride fuel core performed during almost 19 years are very important. Maximum burnup values achieved for different fuel compositions are as follows:

– Plutonium dioxide: 14.1% h.a.;
– Uranium monocarbide: 6.1% h.a.;
– Uranium mononitride: 9% h.a.

Large experience has been gained in reactor operation with failed fuel elements, including fuel elements with deliberately perforated cladding.

In-pile tests of various structural, moderator and absorber materials are carried out on a constant basis in order to study their behavior under irradiation.

Mass transfer of various impurities and nuclides (manganese, cesium, etc.) and their distribution in the primary circuit has been studied.

Methods of control of fission and corrosion products activity in the coolant and on the primary piping walls have been developed.

Sodium corrosion effect on the primary and secondary structural material (Cr18Ni10Ti stainless steel) was studied by taking steel samples (Fig. 1) after 49 000 hours to 210 000 hours operation in sodium at the temperatures from 300 to 450°C.

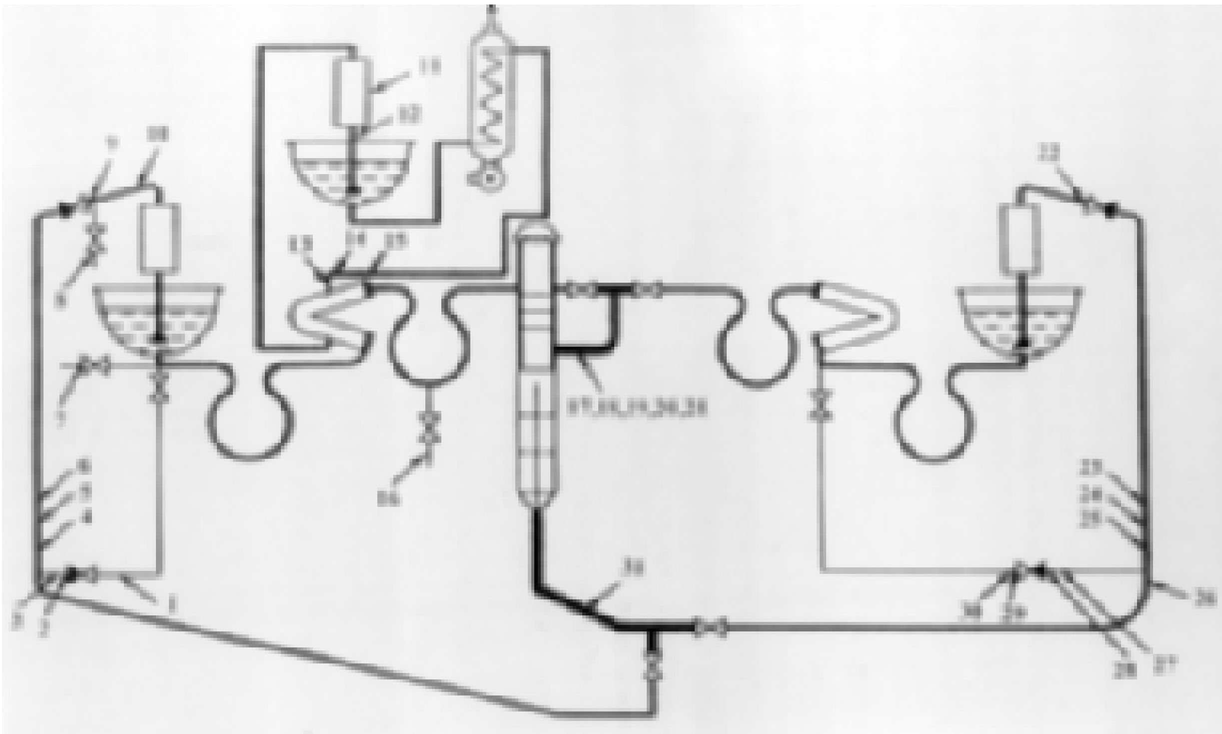


FIG. 1. Points of taking steel samples from the pipes in the primary and secondary circuits in BR-5/BR-10 reactor.

This study revealed no significant corrosion of material used during 40 years in sodium coolant, so it was possible to continue operation of the BR-10 reactor circuits (this study and laboratory tests of structural materials at 500°C have shown that the circuit components made of similar stainless steels can be used in sodium for 60 years without worsening of their properties because of corrosion in sodium, on condition that certain oxygen content in sodium is maintained).

Systems of impurities control and removal from the coolant were developed and optimized. Methods of removal of sodium residues from the circuit after draining have been developed, as well as technology of cleaning of inner surface of the circuit (by steam or steam/gas) from sodium and decontamination of circuits. During reactor operation period, primary circuit cleaning from sodium was carried out three times (in 1961, 1971, and 1980). For this purpose, steam at 130°C temperature was supplied during ~40 hours. Besides, two loops of the primary circuit were subject to decontamination in 1961 and 1971. For instance, in 1971 decontamination procedure repeated 6 times included the following stages:

- Filling with 0.3% water solution of $KMnO_4$ at 60°C for 2-4 hours;
- Washing with distillate at 70°C;
- Filling with 1–1.5% water solution of $H_2C_2O_4$ (oxalic acid) with addition of 1% H_2O_2 (hydrogen peroxide) at 60°C for 2-4 hours;
- Washing with distillate at 70°C.

System of continuous control of the core by delay neutrons has been developed and tested on the reactor, transport time from the core to the sensor being equal to 4 s. In addition, periodic control of activity of gas fission products (xenon and krypton) in the cover gas of the primary pump vessels is carried out. Passive (hydraulically suspended) safety rods were tested in the reactor. Production of isotopes ^{32}P, ^{33}P, ^{35}S, and ^{89}Sr by (n-p) reaction, ^{127}Xe, ^{131}I, and ^{198}Au by (n-γ) reaction; ^{99}Mo by (n-f) reaction, etc. has been organized. Technique has been developed, and now treatment of oncological patients is made on periodic basis using fast neutron beam (total number of treated patients is about 500). Special device has been designed and constructed for producing track membranes having sterilization properties (pore diameter ranging from 0.5 μm to 5 μm, pores density 10^6 to 10^9 L/cm^2). In this device lavsan film is irradiated by the fission products in thermal neutron beam. This filtering film can be used for the final purification of water, separation of form elements from the blood plasma and high-selectivity sterilizing micro filtration of liquids and gases.

Total number of sodium leaks occurred in the BR-5/BR-10 reactor during its operation is 19. These leaks were caused by the following reasons:

i.	Burn-through of the pipeline wall by electric heaters	2 leaks
ii.	Failures of level indicators in the pump vessels	6 leaks
iii.	Failures of sodium valves	7 leaks
iv.	Wrong sequence of procedures of heating pipeline with frozen sodium	2 leaks
v.	Manufacture defect	1 leak
vi.	Crack in the pipeline wall	1 leak

- Pipeline wall burn-through events took place on the early stage of the BR-5 reactor operation, since experience in design and operation of sodium systems at that time was poor causing wrong decisions. These events were caused by the short circuit between electric heater and pipeline resulting in the electric arc. In this view, power supply system of electric heaters was modified, namely: transformer with insulated neutral conductor. Such approach is used in all domestic fast reactors. No more leaks have occurred for this reason.
- Leaks in the sodium valves of the BR-5/BR-10 reactor occurred for two different reasons: failures of sealing bellows (on the early stage of operation), and design defects of the new installed valves (on the later stage). In the first group of events sodium leaks were confined within the guard gland sealing (i.e. the amount of leak was within the few cm^3), and in the second group valve vessel lost its integrity because of thermal expansion of sodium, however, the leaks were also small.
- Heating of circuit with frozen sodium should start from sodium free level, and each heater section should be switched on in the strict sequence only after sodium in the previous section has been molten. If this sequence of procedures is not observed, expansion of sodium caused by phase transition occurs in the restricted volume. This is the cause of sodium communication ruptures. This was caused by either incorrect arrangement of heaters, or errors in operating instructions, or errors of personnel.
- Leak caused by manufacture defect occurred just after cold trap has been installed in the primary circuit. Sodium leaked through the micro crack in the weld on the nozzle of the trap as a result of sodium heating in the pipeline. The cause of the leak was improper control of welds after installation of the cold trap. Such leaks are revealed just after putting into operation of the failed element of the system.
- One leak in the BR-5/BR-10 reactor took place through the crack in the area of drain pipeline connection to the main pipeline. This was caused by insufficient self-compensation of pipeline thermal expansion. Sodium leak was within a few dozens of cm^3. No sodium release beyond thermal insulation occurred. Visual inspection revealed the crack in the body of the drain pipeline connection unit. The crack of ~0.5 mm width spread to the half of pipeline diameter (40 mm). In the vicinity of through crack, on the inner surface of connection unit, there were micro cracks of up to 0.4 mm depth. The leak was caused by limited freedom of drain pipeline movement when the temperature in the main circuit changed.
- Sodium level indicator sensors in the pumps of the BR-5/BR-10 reactor are made of 20 mm diameter stainless steel tube having 0.2 mm wall thickness. These tubes consist of sections with welded current collectors. Leaks occurred through the cracks in the welds of the sections. After welding technology was modified, no leaks in the level indicator sensors took place.

It should be noted that almost all sodium leaks occurred on the early stage of reactor facility operation, i.e. when sodium coolant technology had not been perfectly mastered and new design approaches were still required on the components. No sodium leaks have occurred since 1986. Now steel samples cut off from reactor main vessel and guard vessel dismantled in 1980 are studied in the hot cells of the IPPE. Extremely high neutron fluence has been gained by the reactor vessel (Cr18Ni9Ti steel), namely: 8×10^{22} n/cm^2 exceeding that reached not only by any other reactor vessel, but also by non-replaceable in-vessel structures. Neutron fluence is different over the reactor vessel height ranging from zero to maximum value, thus making it possible to determine mechanical characteristics, microstructure and swelling rate of Cr18Ni9Ti steel as a function of neutron fluence and operating temperature.

2.1.4. Current state of the BR-10 reactor

In the recent years, maximum operating power level of the reactor is 6 MW with permitted maximum fuel burn-up 9% h.a. Fast neutron fluence 6.31×10^{22} n/cm^2 (E_n > 0.1 MeV) has been reached on the reactor vessel, design value being equal to 7.0×10^{22} n/cm^2.

License has been issued for reactor lifetime extension up to 31 December 2002. Experimental studies on methods and technologies of utilization and disposal of reactor components are under way in order to support the project of reactor decommissioning. Special test facility has been constructed for testing methods of sodium waste conversion into the safe condition by solid phase oxidation. Technology of conversion of the primary and secondary cold traps into the safe condition has been developed. Operating experience gained in the BR-5/BR-10 reactor has given the basis for optimization of the principal issues of physics and technology of sodium cooled fast reactors (SFR) and the application of these results for development of further reactor designs: BOR-60, BN-350, BN-600 and BN-800.

2.2. Experimental reactor BOR-60

Main characteristics of BOR-60 reactor can be seen in Table 1.

TABLE 1. MAIN CHARACTERISTICS OF BOR-60 REACTOR

Thermal power	up to 60 MWth
Electric power	12 MWe
Heat supply capacity	20 Gcal/h
Maximum neutron flux	3.7×10^{15} cm^{-2}s^{-1}
Maximum power density in the core	1100 kW/L
Average neutron energy	0.45 MeV
Fuel	UO_2 or UO_2-PuO_2
Enrichment with ^{235}U	45-90%
Maximum Pu content	up to 40%
Enrichment with ^{239}Pu	up to 70%
Fuel burn-up rate	6%/year
Annual neutron fluence	5×10^{22} cm^{-2}
Power rate non-uniformity factors:	
Axial	1.14
Radial	1.15
Volumetric	1.31
Number of cells in the reactor	256
Number of control and safety rods:	
Automatic control rods	2
Shim rods	2
Safety rods	3
Neutron absorber material	B_4C
Coolant	sodium

Cont'd

Coolant temperature:	
at the reactor inlet	330°C
at the reactor outlet	530°C
Sodium flow rate in the reactor	up to 1200 m^3/h
Sodium velocity in the core	up to 8 m/s
Sodium flow rate in two secondary loops	up to 1600 m^3/h
Steam pressure in the third circuit	10 MPa
Superheated steam temperature	500°C
Thermal power of one steam generator	30 MW
Power of sodium-air heat exchanger	30 MW
Reactor run duration	90 days
Run-to-run interval	45 days

Main stages of BOR-60 reactor operation and performance can be seen in Table 2.

TABLE 2. MAIN STAGES OF BOR-60 REACTOR OPERATION AND PERFORMANCE

Stage	Date of issue of Governmental Decree on Design
Development of BOR 60 reactor facility	08 September 1964
Start of design stage	1964
Start of construction	1965
Reactor first criticality	1968
Connection to the grid	1969
Start-up of heat supply facility	1991
Production indices:	
Heat	7.4×10^9 kWh
Electricity	1.2×10^9 kWh
Heat supplied for RIAR site needs	5.6×10^5 Gcal
Reactor availability (for all operating period)	0.64÷0.73

Analysis of failures of BOR-60 reactor is shown in Table 3.

TABLE 3. ANALYSIS OF FAILURES OF BOR-60 REACTOR

Year	Control and safety system	Electric equipment and power supply	Mechanical components	Instrumentation	Personnel errors
1970	7	4	-	1	9
1971	4	5	5	-	6
1972	1	1	3	2	3
1973	4	3	6	4	4
1974	1	4	-	-	1
1975	-	1	6	-	2
1976	-	2	1	-	-
1977	1	6	1	1	2
1978	-	3	1	-	2
1979	1	2	1	-	3
1980	-	-	4	-	3
1981	1	3	3	-	3
1982	-	4	3	-	1
1983	-	1	1	-	1
1984	1	1	3	1	1
1985	2	-	1	-	2
1986	2	-	2	-	1
1987	2	-	1	1	1
1988	1	2	-	2	-
1989	-	1	-	-	-
1990	-	1	-	-	-
1991	-	1	1	1	1
1992	-	1	-	-	-
1993	-	4	-	1	-
1994	-	-	-	-	-
1995	-	-	-	-	-
1996	-	1	-	-	1
1997	-	-	-	-	-
1998	-	1	1	-	-
Total	28	52	44	14	47

- Most of reactor shutdowns were caused by the failures of power supply system;
- Introduction of new equipment results in the increase of number of failures;
  - 1980 – 1998: replacement of feed water pumps;
  - 1985 – 1987: replacement of devices of control and safety system;
- Growth of personnel skill led to considerable decrease of the number of reactor shutdowns caused by personnel errors.

2.2.1. Experience gained in operation of sodium circuit components

2.2.1.1. Reactor and in-vessel devices (IVD)

All IVD including control rod drives operate quite reliably. However, in the early stage of reactor operation seizure of the large rotating plug occurred. Annular gap between reactor vessel and the plug is 4 mm on average. Because of ellipticity of the plug (revealed during its installation), this gap width varies from 3 to 8 mm. No seizures have occurred on the small rotating plug. Study of this phenomenon showed that the cause of seizure was deposits of sodium aerosols on the cold surface of the reactor vessel above the liquid sodium level.

In order to restore free rotation of the plug, some techniques were worked out:

- After reactor shutdown, air flow rate provided for cooling biological shielding and reactor vessel is decreased down to 20%;
- Sodium temperature at the reactor inlet ~300°C is maintained;
- Alloy is heated in the freezing seals of rotating plugs;
- Sodium level is increased up to maximum value, and after 3÷4 hours decreased down to minimum value required for reactor refueling. Difference of levels is 1500 mm.

The procedure is repeated several times, i.e. aerosols are dissolved in sodium. Owing to this technique plug seizure was eliminated.

2.2.1.2. Main primary and secondary pumps

In 1973, one primary pump was replaced because of its high vibration. Studies revealed deformation of the pump shaft (shaft deflection was 1 mm) caused by incorrect technology of its heat treatment after the shaft components were welded together. No failures have been detected in the rest of the pumps. Maximum operation time of the pumps is ~180 thousand hours.

2.2.1.3. Primary and secondary valves

Total number of valves in the reactor facility is 77. Number of “open-close” cycles varies from 20 to 600. Removable part of only one valve having 100 mm inner diameter has been replaced during reactor operation because of loss of integrity of the bellows (without sodium leak to the room).

2.2.1.4. Results of steam generator tests

Steam generator PGN-200M (model of the BN-600 reactor steam generator) has been in operation generating steam for 15 160 hours. It was decommissioned because of inter-circuit leak. This leak was revealed by detectors of hydrogen in the argon cover gas of pressure compensatory tank. Studies have revealed ingress of ~3 kg of water into the secondary sodium and ~200 g of sodium into water-steam circuit. The leak occurred in the joint of one tube to the upper tube plate.

In 1979, tests were carried out in the micro-modular steam generator SG-1 to study sodium-water interaction processes in case of small (up to 0.2 g/s) and large (up to 0.25 kg/s) water leaks. The main results of tests:

- Various methods of leak detection have been verified;
- Serviceability and reliability of SG safety systems have been verified;
- Experience has been gained in accident elimination at BOR-60 reactor.

One-month period was required for SG repair before it was put into operation. Currently, two reverse type steam generators (OPG-1 and OPG-2) are in operation in BOR-60 reactor.

Characteristics of completely tested BOR-60 steam generators are shown in Table 4, and characteristics of OPG-1 and OPG-2 steam generators are shown in Table 5.

TABLE 4. STEAM GENERATOR OPERATING EXPERIENCE - CHARACTERISTICS OF COMPLETELY TESTED BOR-60 STEAM GENERATORS

	Integral SG with helical tubes		Micro-modular direct (SG-1)		Sectional PGN-200M (SG-2) - model of BN-600 SG	
Characteristics	Test modes		Test modes		Test modes	
	basic	max power mode	basic	max power mode	basic	max power mode
Date of putting into operation	1971-1974		1973-1980		1978-1982	
Test duration, h	18 000		32 000		15 000	
Thermal power, MW	20	31	24	30	20	25
Steam capacity, t/h	30	45	35	44	30	36
Feed water temperature, °C	190	210	200	210	200	195
Superheated steam temperature, °C	430	465	450	445	450	470
Steam pressure, bar	90	90	80	80	90	86
Sodium flow rate, t/h	410	530	360	485	400	415
Inlet sodium temperature, °C	430	485	465	455	450	470
Outlet sodium temperature, °C	295	295	280	290	300	300
Average heat flux, MW/m^2	0.180	0.263	0.141	0.175	0.096	0.12
Maximum heat flux, MW/m^2	0.35	0.63	0.50	0.55	1.0	0.98

TABLE 5. CHARACTERISTICS OF OPG-1 and OPG-2 STEAM GENERATORS

Characteristics	Micro-modular reversed type steam generator OPG-1	Modular reversed type steam generator OPG-2
Date of putting into operation	1981	1991
Test duration, h	91 000	43 000
Thermal power, MW	28	26.3
Steam capacity, t/h	42	39
Feed water temperature,°C	210	210
Superheated steam temperature, °C	475	460
Steam pressure, bar	90	90
Sodium flow rate, t/h	400	430
Inlet sodium temperature, °C	480	480
Outlet sodium temperature, °C	280	283
Weight, t	28.5	9.1
Average heat flux, MW/m^2	0.160	0.190
Max heat flux, MW/m^2	0.450	0.550

Reversed steam generator design implies sodium flowing inside the tubes with water-steam flowing on the shell side.

2.2.4. OPG-1 Design

Steam generator consists of eight sections, each one containing preheater, evaporator and superheater modules. It is enclosed in casing with thermal insulation and electric heaters. Module vessel diameter and wall thickness are 194 and 16 mm, respectively. Module tube bundle consists of 19 tubes ∅ 25×3 mm (3.5 mm — thickness of superheater tube wall) spaced at 28.5 mm. Spacing of the tube bundle is assured by the grids. Straight section of the tube bundle is covered by hexagonal guide duct. Figure 2 shows OPG-1 section (one out of eight sections) (2a), and cross-section of OPG-1 module (2b), respectively.

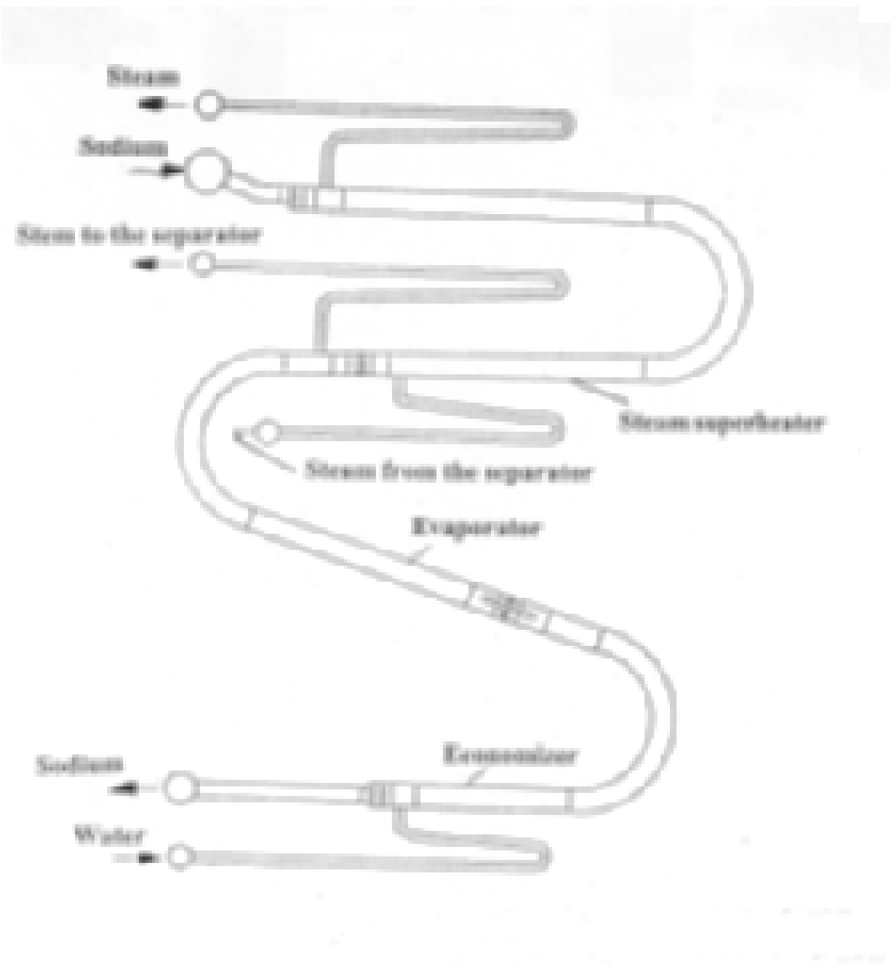


FIG. 2a. OPG-1 section (one out of eight sections).

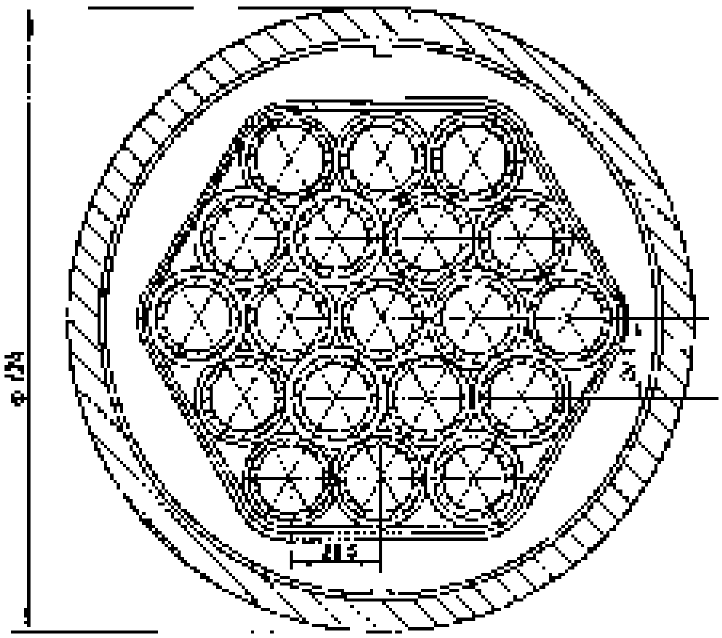


FIG. 2b. Cross-section of OPG-1 module.

2.2.5. Tests of fuel, absorber and structural materials

In the course of reactor operation, big variety of fuel and structural materials (SA wrapper and fuel element cladding materials) have been tested.

The main trends of radiation tests

I Fuel and fuel elements

Sintered ceramic fuel:
- UO_2, $UPuO_2$, (514 SA);
- UC, UN, UPuC, UPuN, UPuCN (18 SA);

Vibropacked ceramic fuel:
- $UPuO_2$ (442 SA);
- UO_2 (210 SA);

Metallic fuel:
- U, Uzr, UPuZr (22 SA);

Cermet fuel:
- $U\text{-}UO_2$, $U\text{-}PuO_2$ (16 SA);

Composite fuel:
- (UpuZr)C (5 SA)

II Absorber materials:

Control and safety rods:
- CrB_2, B_4C, Eu_2O_3, $Eu_2O_3 + H_2Zr$;

Material samples:
- Ta, Hf, Dy, Sm, Gd, AlB_6, AlB_{12}, Eu_2O_3.

III Structural materials:

- Stainless steels;
- High nickel alloys;
- Refractory materials;
- Zirconium alloys;
- Graphite

IV Electromechanical and other materials:

- Electric insulation materials;
- Magnetic materials;
- Special ceramics;
- Biological shielding materials

In 1981, reactor core was loaded with fuel elements having vibropacked pins based on reactor grade plutonium. Introduction of getter solved the problem concerning physical and chemical interaction of the fuel and cladding and assured high average fuel burnup (13 ÷ 15% h.a.). Maximum burnup achieved in some pilot fuel elements was 32% h.a.

In 1998 tests of the fuel elements based on weapons grade plutonium started (18 SA). Also, accelerated tests of the following structural materials are under way:

- Steels used for in-vessel devices (IVD) of water cooled water moderated reactors (VVR);
- Zirconium alloys for VVR core;
- Vanadium based alloys in lithium environment for fusion reactors.

Currently, design and manufacture work is under way on independent loop for in-pile tests of the fuel elements intended for BREST-300 lead cooled reactor.

Strontium-89 production from yttrium targets continues for fabrication of strontium-89 preparation. Also, gadolinium-153 is produced from 'European' targets for fabrication of sources and preparation.

2.2.6. Sodium technology

Some important problems related to handling of sodium (both non-radioactive and radioactive) have been solved in the course of BOR-60 reactor operation, as follows:

- Sodium purity is maintained by the cold traps (CT) having limited capacity in terms of entrapped impurities. These expensive components required replacement. In order to avoid replacement, system for CT regeneration has been designed, constructed and put into operation at the reactor facility.
- One of the most important tasks is to assure reliable operation of the primary systems and components with simultaneous decrease of personnel dose rates. Now highly effective stationary system for sodium purification from caesium (SSOT) is provided on the reactor facility.
- New technologies have been developed solving the problem of decontamination of sodium components and disposal of sodium residues in the components to be repaired or decommissioned.

During 1995-2001 period, RIAR and OKB GIDROPRESS were involved in the studies on justification of extension of BOR-60 reactor main components lifetime showing that reactor facility can be operated for 20 years more, i.e. its total lifetime can be extended up to 40 years.

By now, permission has been issued for BOR-60 reactor operation until 31st December 2009.

2.3. Beloyarskaya NPP with BN-600 reactor

Main characteristics of BN-600 power unit are shown in Table 6.

TABLE 6. THE MAIN CHARACTERISTICS OF BN-600 POWER UNIT

Reactor:	
- arrangement	pool-type
- support	at the bottom
- vessel cooling agent	cold sodium
- number of heat removal loops	3
- inlet/outlet sodium temperature, °C	377/550
- sodium flow rate, t/h	25 000
Core and fuel:	
- fuel	uranium dioxide pellets
- fuel burn-up max/average, %h.a.	10/6
- diameter, cm	206
- height, cm	104
Intermediate heat exchanger	shell-and-tube design, secondary sodium flowing on the tube side
Primary pump:	
- design	centrifugal, one stage
- rpm	250 – 970
Steam generator:	
- design	once-through, section & modular
- inlet/outlet sodium temperature, °C	(8 sections, 24 modules)
- inlet/outlet water/steam temperature, °C	518/328
- live steam pressure, MPa	241/507
	14
Secondary pump:	
- design	centrifugal, one stage
- rpm	250 – 750
Turbogenerator	standard 210 MW
Decay heat removal system:	
- primary and secondary circuits	normal operation system
- third circuit	steam generator- deaerator
Refueling system	2 rotating plugs, vertical refueling mechanism
Fuel transfer system	elevators with guide ramp
Spent fuel storage	in-vessel storage, sodium and water pools
Washing of subassemblies from sodium	steam-gas-water

2.3.1. The main stages of the BN-600 reactor operation

The main stages of the BN-600 reactor operation are:

- BN-600 power unit was connected to the grid on 8 April 1980;
- Rated power level 600 MWe was achieved on 22 December 1981;
- Since 1982 power unit has been operated to produce electricity and heat on commercial basis;

- The first modification of the reactor core was made in 1987;
- The second modification of the reactor core made in 1993 facilitated reaching design fuel; burnup of 10% h.a.

Work on scheduled replacement of SG evaporator modules and extension of their lifetime has been carried out.

2.3.2. Main results of BN-600 reactor operation

Power unit operation during over 20 years has demonstrated good agreement between design and real characteristics of the main components. The following high technical and economical indices have been achieved:

- Power unit gross efficiency 41.93%;
- Load factor over the whole period of operation 69.9% (over 74% — if early stage of operation is neglected).

By 28 January 2002, total time of BN-600 reactor operation since initial start-up is 150 thousand hours, i.e. 17 years. Operating experience has made it possible to improve technological modes, systems and components of the BN-600 reactor power unit. In particular, the following modes were optimized:

- Connection of heat removal loop to the reactor in operation;
- Reactor decay heat removal;
- Power unit start-up;
- Connection of steam generator section with power unit in operation.

According to the original BN-600 reactor design there were two zones of ^{235}U enrichment (21 and 33%, respectively) in the uranium dioxide core, maximum fuel burnup being equal to 9.7% h.a. Operation of the original BN-600 reactor core revealed that because of small gaps between subassemblies and the wrong choice of SA wrapper and fuel element cladding material fuel burn-up was limited by 7.3% h.a. value. Moreover, even this burn-up caused fuel element failures and considerable bending of subassemblies at the end of each reactor run. In order to eliminate bending of subassemblies and fuel element failures, the first modification of the core was made in 1987. The main features of this modification were addition of the third enrichment zone in the core (17, 21, and 26% enrichment zones, respectively, maximum fuel burnup 8.3% h.a.), elimination of rotation and shuffling of subassemblies, increase of the core height from 75 to 100 cm with decrease of linear power from 54 to 47 kW/m, and application of the new cold worked (~20%) steel as SA wrapper and fuel cladding material. Refueling was made twice a year, with the interval 165 eff. days. As a result of modification, there were no more fuel element failures, and further burnup increase became possible. After the second modification of the core implemented in 1993, 10 and 11.8% h.a. fuel burnup was achieved respectively in the standard and pilot subassemblies. This modification was based on the application of new structural materials, and, in particular, ferritic-martensitic steel was used for the first time in the world practice as structural material for the SA wrapper. Studies on the fuel burnup increase were based on the results of in-pile tests of pilot subassemblies. Total number of irradiated subassemblies was 318 (over 40 thousand fuel elements). Uranium-plutonium vibropacked and pellet fuel subassemblies were tested successfully. In-pile tests of 28 pilot subassemblies (PSA) with MOX fuel pellets have been completed, and 12 more PSAs are now irradiated. Besides, 6 vibropacked PSAs have been irradiated and 3 PSAs now undergo in-pile tests. In general, operation of reactor and in-vessel structures has been successful. The most significant problems encountered during operation refer to sodium composite deposits on the reactor vessel surface above the sodium level and prematurely exhausted lifetime of control rod guide tubes. Sodium composite deposits on the reactor vessel surface in the cover gas plenum were caused by moisture brought by the argon gas supplied for replenishment and air entering reactor vessel in the process of in-vessel replacement. In January 1987, ingress of deposits into sodium occurred during reactor operation at the rated power level. This added small positive reactivity (0.03% $\delta k/k$), which was compensated by automatic control system having considerable margin for compensation of such perturbations. In order to eliminate moisture and air penetration into the reactor, argon replenishment system was modified, drying unit was installed and sealing devices used for in-vessel components replacement were improved.

Since 1995, seizure of the small rotating plug was observed during reactor refueling, and this seizure became stronger with time. In order to determine the cause of this phenomenon, in 1997 a drilling was made of the plug body during scheduled reactor shutdown and its bearing unit was visually inspected. As a result of inspection, sodium was revealed in the bearing unit. An attempt was made to remove the sodium by heating, but it failed. So it was decided to remove sodium and replace bearing (if necessary) during scheduled preventive repair (SPR) in 1998. In 1998, during SPR combined with reactor refueling, sodium was removed and the bearing was replaced. These procedures were carried out after the plug had been moved 650 mm upwards in the gas-tight cask. Replaced bearing was plugged with sodium. Analysis showed that the amount of sodium (~15 kg) corresponded to the rate of sodium vapor transfer in the gap between the large and small rotating plugs.

After that, the plug was lifted up to 2100 mm height for inspection and measurements. Scratches were revealed on the lateral surface of the plug, so the surface was ground. These measures made it possible to restore design plug rotation force. In the early stage of operation, failures of pump shafts and pump-drive shaft couplings occurred causing unscheduled reactor power decrease. These failures were because of coincidence of shaft resonance frequency and that of torsional vibrations caused by rectified rotor current pulsation that could not be eliminated. After source of failure had been identified and pump rotation frequency had been made different from resonance value, no more shaft failures were observed. Later, shaft design was modified, as well as the method of switching pump to non-controlled operation mode when operating at the steady state power level, and so the source of pump failures was eliminated. Reliability of the primary pumps was increased owing to the increase of lifetime of the basic structural elements. The most significant was increase of the impeller lifetime from 35 000 to 50 000 hours. Sectional/modular steam generators are used in the BN-600 reactor. Each one out of three steam generators consists of eight sections. Each section includes three modules, namely: evaporator, superheater and reheater. These modules are connected by the pipelines to form section. Sections of one steam generator are also connected by the pipelines with valves for section isolation on either sodium or water-steam side in case of failure. Owing to this design, failed section can be put out of operation with all the rest sections still in operation without decreasing reactor power. During the whole period of SG operation, 12 leaks of steam and water into sodium have occurred, half of these leaks took place in the first year of operation because of manifestation of hidden manufacture defects. Inter-circuit leaks took place mainly in the superheaters (6 events) and reheaters (5 events), while only one leak occurred in the evaporator (Table 7).

TABLE 7. CHARACTERISTICS OF INTER-CIRCUIT LEAKS IN THE BN-600 REACTOR STEAM GENERATOR MODULES

Leak ordinal № / Leak parameters	1	2	3	4	5	6	7	8	9	10	11	12
1. Modules	RH	SH	RH	SH	SH	SH	SH	SH	E	RH	SH	RH
2. Leak size	L	L	S	S	S	S	L	S	S	S	S	S
3. Date of leak	24.06.80	04.07.80	24.08.80	08.09.80	20.10.80	09.06.81	19.01.82	22.07.83	06.11.84	10.11.84	24.02.85	24.01.91
4. Operation time before leak, hours	1000	968	1145	1454	950	1640	4019	19584	26032	14512	26078	44000
5. Electric power of reactor facility at the time of leak, Mwe	270	65	313	362	332	210	550	606	240	600	400	596
6. Secondary sodium temperature at the SG inlet/outlet, °C	460/300	314/299	465/300	468/298	460/299	401/300	500/301	506/304	510/305	510/305	480/300	513/315
7. Parameters of the third circuit:												
-feed water temperature, °C	156	-	162	163	159	159	164	164	238	240	163	240
- live steam /reheated steam temperature, °C	440/432	-	450/453	461/453	456/447	307/187	490/483	501/493	504/497	506/496	470/462	504/499
- superheater/reheater steam pressure, Mpa	11.2/1.0	-	10.8/0.92	10.3/1.31	11/1.03	5.5/0.36	11.2/2.2	12.1/2.2	12/2.2	12.1/2.2	11.9/2.8	11.9/2.1
8. Time of reaching accident setting, min	-	-	4	5	8	5	2	7	9	-	5	4.5
9. Leak rate, g/s	0.02-6	0.1-0.615	0.09-15	0.2-0.3	0.0064-0.23	140	250	-	0-3	0.02	0.14	4.6
10. Amount of water penetrating into the secondary circuit, kg	40	17.87	7	0.18	0.78	40	20.3	2.77	1.8	0.75	0.73	8.3

Analysis of leak events has demonstrated high resistance of SG design with respect to inter-circuit leaks. Operating experience confirmed the choice of SG concept: decrease of power generation caused by 12 water-to-sodium leaks was as low as 0.3%. Thus, in spite of SG leaks, its sectional/modular design assured planned rate of power increase and high performance. Justification of possibility of evaporator lifetime extension from 50 thousand hours (design value) to 105 thousand hours made in the process of their operation made it possible to replace evaporators only once during the power unit lifetime instead of planned three times. This was the result of evaporator conditions study program implemented for many years, modification of water chemistry, decrease of the number of reactor transients and accidents as compared to predicted number, as well as procedures of chemical cleaning and washing out friable deposits with water performed on periodical basis. Since 90-ies, scheduled replacement of SG evaporator modules has been carried out. During BN-600 reactor operation, 27 sodium leaks have occurred. Below given is sodium leaks distribution with respect to their causes:

i.	Failures of SG sodium valves	5 leaks
ii.	Defects of flange joints	5 leaks
iii.	Wrong sequence of procedures of melting sodium in the pipeline	4 leaks
iv.	Holes made by personnel	2 leaks
v.	Manufacture defects	3 leaks
vi.	Sodium valve defects	2 leaks
vii.	Cracks in the pipelines	6 leaks

- Sodium valves provided for isolation of the BN-600 reactor steam generator modules from the circuit have flange joints backed up by «moustache» welds. It was initially assumed that the flange joints would assure integrity, so insufficient importance was attached to the quality of welds. After sodium leaks had occurred in the flange joints, «moustache» welds were recovered with subsequent quality. Since then no more leaks have occurred in the SG valves.
- In the sodium systems, weld joints are used as a rule. Flange joints are used as an exception in the sodium preparation system. Vessels for sodium transportation (such as railroad tank-cars) are connected to the sodium systems using removable pipeline sections. Leaks in the flange joints occurred very often after connection had been made. These leaks were detected on the early stage by the operator or detection systems.
- Causes of leaks in case of melting frozen sodium in the system are similar to those in the BR-10 reactor.
- Several sodium leaks were caused by personnel actions. For instance, cuts were made on the pipelines for making repair, and then sodium was erroneously supplied to the cut. All these leaks were immediately detected.
- Leaks caused by low quality control of welds after installation were detected immediately after putting into operation of failed system element.
- There was sodium leak in the valve of the BN-600 reactor occurred as a result of wear of seal between the vessel and bellows (less than 1 kg of sodium escaped, and no burning occurred). In the other case, poor quality of weld (faulty fusion and craters) was the cause of leak (the amount of leaked sodium was also within 1 kg, and no burning occurred).
- Six leaks occurred through the cracks appearing in the pipelines during BN-600 reactor operation. The main causes and nature of the events are similar to those in the BR-10 reactor, i.e. wrong design approaches or installation errors causing insufficient self-compensation of pipeline thermal expansion.

Most sodium leaks out of 27 that occurred during BN-600 reactor operation were small leaks: in 21 cases the amount of released sodium was within 10 L (ranging from 0.1 to 10 L). In the rest 6 cases the amount of released sodium was 30, 50, 300, 600, 650 and 1000 L. Table 8 gives the main characteristics of large sodium leaks (over 10 L).

TABLE 8. THE MAIN FEATURES OF LARGE SODIUM LEAKS IN THE BN-600 REACTOR

Date of leak	Leak location	Detection method	Cause of the leak	Amount of sodium released, kg
13.01.1980	Sodium receipt system	Smoke ionization detectors	Flange joint defect	50
11.08.1981	Sealing of SG valve	Electric heater control, ionization detectors	Flange joint defect	300
02.07.1982	Sealing of SG valve	Visual inspection by personnel	Flange joint defect	30
31.12.1990	SG drain pipeline	Electric heaters	Manufacture defect	600
07.10.1993	Primary sodium purification system	Electric heaters, control of radioactive aerosol appearance	Insufficient pipeline self-compensation	1000
06.05.1994	IHX drain pipeline	Visual inspection by personnel	Cutting pipeline before sodium is frozen	650

Sodium fire was observed in 14 cases. All leaks were detected in due time by control systems or operators. Total number of leaks can be distributed with respect to reactor facility components in the following way:

- Sodium reception system 5
- Out-off valves of SG modules 5
- Auxiliary secondary systems 12
- Auxiliary primary systems 5

Powders were used for confining and extinguishing non-radioactive sodium fires. It was only in the case of large leak and fire of radioactive sodium from the primary circuit, that design algorithm of confinement of sodium fire consequences was implemented successfully: in this case radioactivity release (10.7 Ci) was well below permissible limit. There was no need in using drainage based fire fighting systems. It was leak occurred on 07 October 1993 in the auxiliary pipeline of 48 mm diameter of the primary sodium purification system, that was the most serious abnormal operation event for the whole period of power unit operation. This event refers to the 1st level according to «on-site impact» and «defense in depth degradation» parameters. This resulted in insignificant radioactivity release through ventilation stack, which was equivalent to 0.001 of natural background at the NPP control area boundary.

Thus, abnormal operation events occurred in the power unit resulted in no radiation impact on residents and environment, since all these events were below the International Nuclear Event Scale (according to «off-site impact»), i.e. could be neglected.

In total, 104 abnormal operation events have occurred during power unit operation (as of December 2001) resulting in the unscheduled reactor power decrease. Distribution of these events with respect to years, components (irrespective of the cause) and causes, respectively, is shown in Figs 4 – 6.

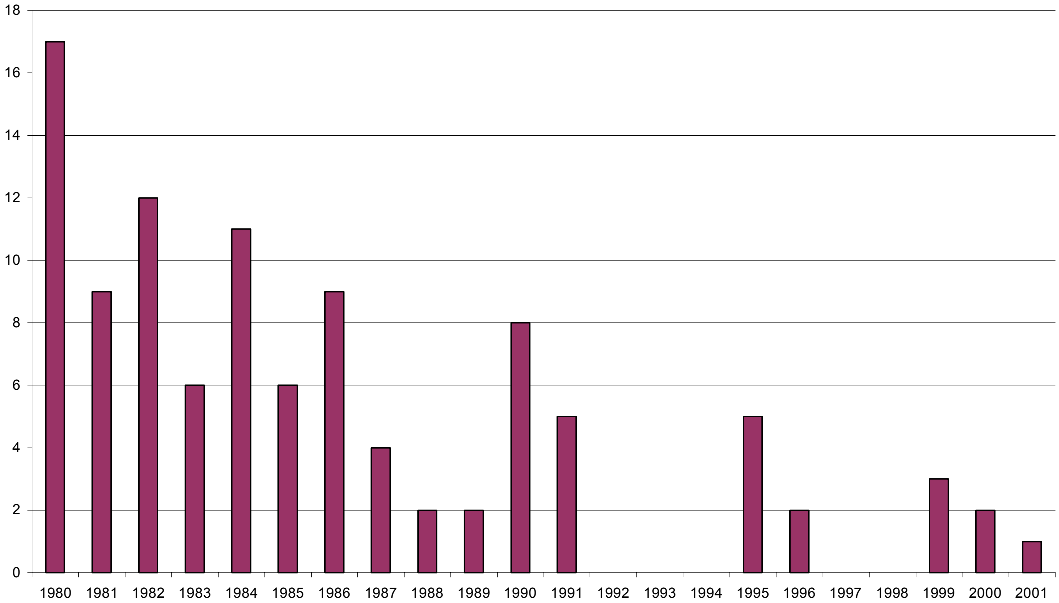


FIG. 4. Number of abnormal operation events causing power decrease.

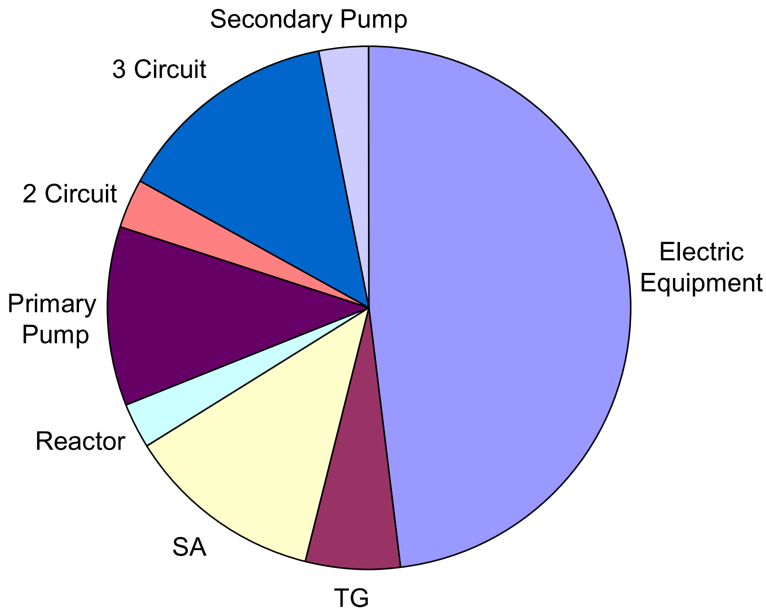


FIG. 5. Abnormal operation events caused by different component failures.

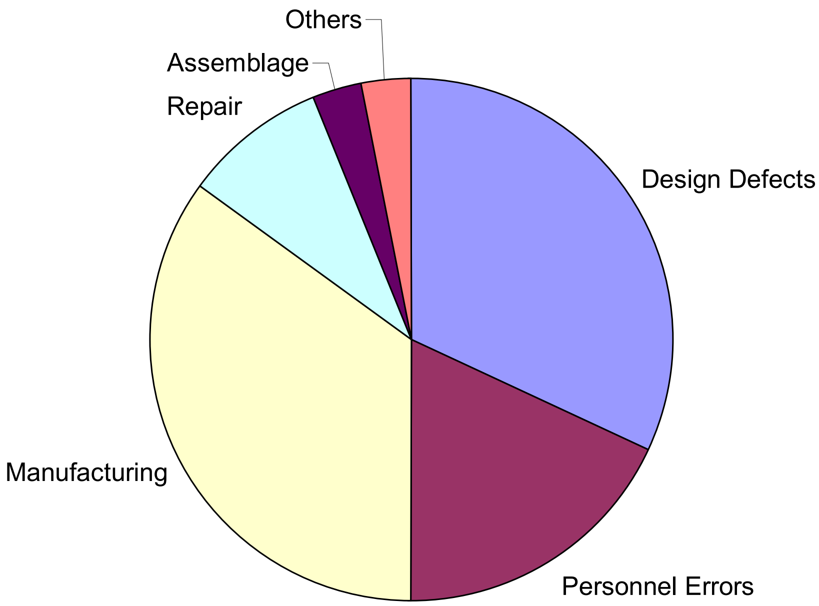


FIG. 6. Causes of abnormal operation events.

Study of the BN-600 reactor power unit operation experience gained for over 20 years has shown the possibility of fast reactor application for commercial production of electric energy.

2.3.3. Current state of BN-600 reactor

Power unit now operates in stable mode on the rated power level. Only in the winter, when heat supply to the consumers increases, some decrease of electricity production is observed.

During 2001, 4199.2 million kWh energy was produced corresponding to design requirements. Load factor is 79.9%. No unscheduled shutdowns of power unit occurred last year, only one case of one loop shutdown took place. Activity is now under way on extension of power unit lifetime for 10 years more, i.e. until 2020.

2.4. Participation in activities on BN-350 reactor decommissioning

BN-350 reactor was designed by the Russian organizations, and Russian specialists participated in its operation and power mastering. Valuable experience of BN-350 reactor operation has been taken into account in BN-600 and BN-800 reactor designs.

Therefore it is reasonable to involve Russian organizations: VNIPIET, SSC RF IPPE and OKBM in the development of basic design of BN-350 reactor decommissioning, and first of all, in the work related to management of the primary and secondary sodium (purification from cesium, circuits draining and sodium residues removal from non-drained sections and conversion to chemically inert condition), liquid and solid radwaste, cleaning and decontamination of the components, and radiation control and safety measures in all stages of the BN-350 reactor decommissioning.

2.5. Summary of the main results of FR operation in Russia

In Russia, experience of successful operation of sodium cooled fast reactors (SFR) has been gained, namely ~125 reactor-years (taking into account BN-350 reactor).

This experience covers almost all stages of life of reactor facilities from design and R&D work for design justification up to reactor decommissioning. It should be noted that there is no one decommissioned fast reactor in Russia, however some practical experience has been gained in this area. This experience is mainly based on utilization of some components of reactor plant as well as work on BR-10 reactor updating. Besides, current activity within the framework of BR-10 preparation for decommissioning is also of great interest.

As regards aspect of fire danger related to sodium coolant, it should be noted that about one third of all 61 sodium leaks occurred in all reactors (including BN-350) were caused by the errors of operators or repair personnel. About half of the leaks occurred during repair work, in the start-up and adjustment stage or in the system for sodium coolant preparation (which could not impact reactor safety). No one of the leaks has imposed threat to reactor safety.

Experience has been gained also in operation of various steam generator designs. Based on this experience, effective automatic SG safety systems have been designed to assure shut-down of failed SA on the stage of «small» leak (before it grows to «large»). Thus, any impact on the reactor plant from failed SG is eliminated.

The following conclusions can be drawn on the basis of experience analysis:

- Feasibility of technology of sodium cooled fast reactors has been confirmed in all stages of their life;
- Successful, reliable operation of the BN-600 reactor has demonstrated possibility of SFR application as cost effective source of electric energy assuring high load factor;
- Practical safety of SFR has been demonstrated (analysis of sodium leaks occurred has revealed no catastrophic consequences).

3. STATE-OF-THE-ART IN PRESERVATION OF EXPERIENCE GAINED IN FAST REACTOR AREA

Why has necessity of experience preservation appeared?

The overwhelming majority of the most experienced and skilled specialists who have made the main contribution to formation and development of SFR in Russia are currently retired. These specialists have gained vast practical experience in fast reactor operation and technology, as well as in the area of design and justification of design approaches. In view of some pause in fast reactor development and change of generations, the problem preservation of experience gained in SFR area has arisen.

This experience has been recorded in the large number of various documents (articles, papers, reports, design documents, surveys, recommendations, etc.). Therefore, as regards its preservation, it means first of all work on experience systematization, classification and generalization.

Not only experience in fast reactor operation and decommissioning should be preserved, but also that gained in reactor design including results of analytical and experimental studies made within the framework of R&D activities (both in-pile and out-of-pile studies) for justification of fast reactors.

3.1. Current state and problems

Work on experience preservation is under way in the organizations and institutes involved in SFR development.

Within the framework of the contracts including those with international partners some studies have been made on summarizing experience gained in some topics concerning:

- Operation of the BN-600 reactor (design approaches, SA bending, SG, reactor operation with failed fuel elements, transients and sodium leaks);
- Operation of uranium mononitride core in the BR-10 reactor;
- Results of post-irradiation studies of fuel elements with cladding made of ChS-68 steel irradiated in BOR-60, BN-350, BN-600, etc.

Electronic version of bibliographical catalogue of publications on SFR issued with participation of IPPE from 1952 to 1998 has been developed.

However, because of the lack of investments, work on preservation and generalization of experience gained in the area of SFR has been decreased.

There are many enterprises and institutes participating in the work on SFR, such as:

- State Scientific Center of the Russian Federation – Institute for Physics and Power Engineering (SSC RF IPPE), Obninsk;
- Experimental Machine Building Design Bureau (OKBM), Nizhny Novgorod;
- St. Petersburg Research an Design Institute ATOMENERGOPROEKT (SPAEP), St. Petersburg;
- State Research Center of the Russian Federation — Research Institute of Atomic Reactors (SRC RF- RIAR), Dimitrovgrad;
- Experimental Design Organization 'Gidropress' (OKB GP), Podolsk;
- Beloyarskaya NPP (BNPP), Zarechny, etc.

Considerable experience has been gained in these organizations in design, construction and operation of SFR.

4. PROPOSALS ON APPROACHES TO ORGANIZATION OF CO-OPERATION

On the previous TWGFR meeting in Kazakhstan, it was recommended to this IAEA meeting to include into the agenda (in addition to discussion on the results of fast reactor operation and decommissioning) the issue of possibility of organization of joint work on preservation and generalization of experience gained in operation of SFRs.

4.1. Common nature of the problem of FR experience preservation

By now, initial stage of SFR development has been completed. The main result of this stage is development and improvement of basic engineering and technological approaches practically confirming feasibility of commercial SFRs producing electricity. Obviously, the next stage of development of SFRs, i.e. large-scale construction of commercial power units will not start before 2020. Thus, the problem of preservation and generalization of experience gained in the area of SFRs including operation experience exists in each country, where this trend of nuclear power has been and is still developed. In each country, efforts are made to solve this problem using various methods.

It should be noted that all these countries should be interested in preservation of this experience gained in each country. This is because experience of each country is a unique contribution to that gained in other countries. Only the whole experience can give complete idea of the problems related to SFRs.

Importance and urgent nature of the task of preservation of FR operating experience is also caused by the fact that fast reactor plants have been/are decommissioned in many countries. Majority of personnel of these plants have been either retired or left for other areas. Thus, experience gained in SFR operation has been mainly recorded only in some certain documents, namely: reports, instructions for operators, design and working materials, conclusions, surveys, etc.

4.2. Approaches to the problem solution

The task of preservation of operating experience on SFRs is a versatile, complicated problem. Without pretending to get comprehensive solution of this problem, systematization of documents containing data should be made as an initial step. Only availability of complete catalogues of such documents on each aspect of SFR experience will make it possible to turn to the next stage, namely: generalization of the experience and development of recommendations for the advanced SFR designs.

4.3. Possible contribution by TWGFR

One of the most important directions of activity of IAEA Technical Working Group on Fast Reactors (TWGFR) is assistance in the work carried out in different countries on preservation of experience gained in SFR area (including operation experience).

It should be noted that TWGFR participates actively in this work on a permanent basis. Its functions are as follows:

- Holding meetings on exchange experience gained in SFR operation and decommissioning;
- Preparation and issue of various materials summarizing experience gained in either specific or most general aspects of SFR technology.

For instance, there is SFR design characteristics database which is updated on a periodic basis, and also periodic reports on the status of development of SFR technology. In these publications, there are rather extensive lists of references on related topics. However these lists of references cannot be exhaustive, since there are, as a rule, only open publications (papers, articles, books, etc.).

From this standpoint, assistance by TWGFR in the development of comprehensive catalogues of materials on all aspects of SFR operation and decommissioning experience issued in all countries would be quite significant and useful contribution to this experience preservation and generalization.

However, this work implementation is obviously impossible within the framework of a meeting. In this view, proposal has appeared to study the possibility of organization of the Coordinated Research Project (CRP) on this issue.

4.4. Proposals on possible goals and tasks of CRP

There is no doubt that creation of such catalogues of materials containing information on SFR operating experience is hard and expensive work, since a large number of specialists from different institutions should be involved.

So, what parts and directions of this work could be implemented within the framework of the Coordinated Research Project?

First of all, TWGFR could coordinate assistance in working out common approach to formation of such catalogues in TWGFR member states. Such catalogue can be made either as paper document, or electronic database. In the former case, its value is higher because of the possibility of on-line processing of large arrays of data using various criteria.

Titles of the documents containing data on the SFR operating experience (with attached abstracts and other document attributes) are used as input data in catalogues. Obviously, it is expedient to classify the documents in accordance with topic headings corresponding to different aspects of SFR operating and decommissioning experience and facilities. Topic headings, in turn, can be divided into subheadings.

In order to work out common concept of catalogue formation the following subjects are proposed to develop and agree upon with work participants:

- Classification on topic headings (subheadings);
- Format of data presentation (list and form of their attributes);
- Lists of key words and terms typical for each heading (subheading);
- List of characteristics taken into account in the catalogue of SFR plants and their classification;
- List of acronyms and measurement units;
- Language of data presentation (English).

Besides the title of the document, the following attributes can be used:

- Index of headings (subheadings) this document refers to;
- Index of the plant (plants) described or mentioned in the document;
- Authors and their affiliations;
- Country;
- Year of document issuing;
- Place of storage (publication);
- List of key words used in the document;
- Abstract, i.e. brief description of contents of the document, etc.

Proposed project of formation of the catalogue implies of course the possibility of its correction in accordance with the comments and recommendations made by work participants.

Thus, a document will be issued as a result of CRP with the following items developed and agreed upon with the project participants:

- List and structure of topical headings (subheadings);
- Data presentation format (list and form of their attributes);
- Lists of key words and terms typical for each heading (subheading);
- List of acronyms and measurement units;
- List of characteristics included in the catalogue of SFR plants and their classification.

The scope and nature of the proposed work make it possible to reliably implement it within the framework of Coordinate Research Project.

This is the necessary preparation stage for development of the common unified catalogue of materials on experience of SFR operation and decommissioning provided the work is financed. This work is valuable and promising even if separate catalogues are created in TWGFR member states, since all these catalogues meet certain standard requirements agreed upon by all states participating in the work.

Such databases would be useful, because all interested states could address the owners of material of their interest in order to acquire or exchange information.

Upon successful working out of common structure and format for catalogue of materials on experience of SFR operation and decommissioning similar work can be implemented with respect to the catalogue of materials on general experience gained in SFR area including not only operation, but also design and R&D issues, etc.

FAST REACTOR DECOMMISSIONING EXPERIENCE

(Session 2)

DECOMMISSIONING EXPERIENCE FROM THE EXPERIMENTAL BREEDER REACTOR-II

S.P. HENSLEE, K.E. ROSENBERG
Argonne National Laboratory-West (ANL-W), Idaho Falls, USA

Abstract

University of Chicago as Operator of Argonne National Laboratory (AArgonne@) under contract No. W-31-109-ENG-38 with the U.S. Department of Energy. The U.S. Government retains for itself, and others acting on its behalf, a paid-up nonexclusive, irrevocable worldwide license in said article to reproduce, prepare derivative works, distribute copies to the public, and perform publicly and display publicly, by or on behalf of the Government.

1. INTRODUCTION

Consistent with the intent of this International Atomic Energy Agency technical meeting, decommissioning operating experience and contributions to the preparation for the Coordinated Research Project from Experimental Breeder Reactor-II activities will be discussed. This paper will review aspects of the decommissioning activities of the Experimental Breeder Reactor-II, make recommendations for future decommissioning activities and reactor system designs and discuss relevant areas of potential research and development.

2. EBR-II OPERATING SUMMARY

The Experimental Breeder Reactor-II (EBR-II) was designed as a 62.5 MWt, metal fueled, pool reactor with a conventional 19 MWe power plant. The productive life of the EBR-II began with first operations in 1964. Demonstration of the fast reactor fuel cycle, serving as an irradiation facility, demonstration of fast reactor passive safety and lastly, was well on its way to close the fast breeder fuel cycle for the second time when the Integral Fast Reactor program was prematurely ended in October 1994 with the shutdown of the EBR-II.

The shutdown of the EBR-II was dictated without an associated planning phase that would have provided a smooth transition to shutdown. Argonne National Laboratory and the U.S. Department of Energy arrived at a logical plan and sequence for closure activities. The decommissioning activities as described herein fall into in three distinct phases.

2.1. Current status

The plan describing the final condition of the EBR-II, was implemented in October 2000. By this time many important steps had been performed, forming the basis for the plan. Specifically, defueling of the reactor, the design and construction of the sodium process facility and initial sodium processing had taken place.

At the writing of this paper, all physical work associated with the decommissioning activities of the EBR-II are complete. Remaining activities are limited to completion of final documentation specifying actions taken on a system-by-system basis, providing the final configuration control. The following activities have been completed:

— Reactor defueling;
— Spent fuel placed in interim storage;
— Sodium Process Facility made operational;
— Primary and secondary sodium coolant removed from reactor systems;
— Sodium potassium coolants processed with sodium;
— Sodium coolant converted to solid sodium hydroxide for disposal;
— Passivation of sodium remaining within secondary and primary systems; and
— Reactor and non-reactor systems placed in a radiological and industrially safe condition.

2.2. Phase I EBR-II decommissioning activities

The initial phase of decommissioning activities was reactor defueling, completed in December 1996. Defueling was initiated in October 1994 and was completed 14 months later, 3 months ahead of schedule. Defueling included:

— Removal of 637 assemblies from the reactor core;
— Washing sodium coolant from and drying each assembly;
— Transfer of assemblies to a hot-cell facility for disassembly and repackaging; and
— Transfer of spent fuel to interim storage.

Argonne originally anticipated significant modifications to both fuel handling equipment and control systems prior to initiation of defueling. However, in order to minimize the time required to defuel, the risk of potential fuel handling equipment failure was accepted. Fuel handling equipment performed well with only routine maintenance providing an adequate safety margin during nearly continuous operation. Treatment of the EBR-II spent fuel is a state-of-the-art electrometallurgical application and will not be discussed here.

Performed in parallel to reactor defueling was the design and construction of the Sodium Process Facility. The design and construction of the Sodium Process Facility is an application of known and demonstrated technology. The conversion of alkali metal to alkali hydroxide compounds by reaction with alkali hydroxide is routinely performed around the world, e.g. DFR/PFR, at Dounreay and as planned at Superphénix.

2.3. Phase II EBR-II decommissioning activities

The second decommissioning phase was sodium coolant removal and reaction. Construction of the Sodium Process Facility provided Argonne the ability to react elemental sodium with water to form sodium hydroxide, at compositions ranging from 30 to 73% by weight. All sodium treated as part of the decommissioning activities was reacted to a 73% by weight sodium hydroxide. Compositions of 70% by weight sodium hydroxide or greater are solid below 60°C. The U.S. Environmental Protection Agency, the regulatory agency for reactive and corrosive elements and compounds, does not regulate solid sodium hydroxide allowing its disposable as a low-level waste.

By late summer 2000 the secondary sodium coolant had been reacted to a solid sodium hydroxide waste and the primary sodium was being drained. Processing of the primary sodium was completed in March 2001. In addition to the sodium utilized as EBR-II coolant, 372 tonnes. Argonne also stored the Fermi I primary sodium inventory 281 tonnes, that was also reacted to sodium hydroxide and disposed of as low-level waste. A total of 653 tonnes of sodium was converted to 1 450 tonnes of solid sodium hydroxide.

2.4. Phase III decommissioning of the EBR-II

The third and final phase of the decommissioning activity was the placement of reactor and non-reactor systems in a radiological and industrially safe condition. The planning for this phase included a detailed system-by-system evaluation to determine necessary actions based on the following definition:

Radiologically and industrially safe is the placement of equipment and facility in a condition that does not pose any unusual, unexpected or additional industrial safety risk and does not pose a radiation or contamination risk beyond normal EBR-II levels for controlled access areas.

As the system-based planning was developed, necessary or newly identified surveillance activities were identified. Surveillance provides the regulator assurance that EBR-II systems would not deteriorate.

A major component of the radiologically and industrially safe strategy was treatment of the residual radioactive sodium remaining within system piping and components. The conversion of the exposed residual sodium surfaces to a non-reactive layer of sodium carbonate was accomplished through a process called passivation. Passivation (or carbonization) is achieved through the controlled humidification of the carbon dioxide cover gas. The presence of water vapor in carbon dioxide forms a layer of sodium carbonate (sodium bicarbonate). The carbonate layer remains porous and does not impede reaction rates until a layer of greater than 20 cm has been achieved. All exposed surfaces of sodium remaining within both the primary and secondary systems have been passivated.

Subsequent to the draining of the primary sodium a close circuit TV camera was inserted into the EBR-II primary tank. Among the objectives of this visual inspection were: confirmation of completion of the sodium coolant draining process, provide a visual understanding of the draining process and residual sodium deposition locations and finally provide a visual assessment of tank and component integrity.

Visual examination provided evidence of the draining process, confirming the removal of all but very small quantities of residual sodium. An additional visual examination followed the completion of passivation of all primary systems.

3. SUMMARY AND RECOMMENDATIONS

Recommendations for future decommissioning activities, reactor system designs and identification of areas of potential research and development are the result of lessons learned from EBR-II decommissioning activities. Although many of the recommendations are not new, they were included due to the significance to this discussion.

Reactors plants should be shutdown in accordance to detailed planning allowing operators and regulators a common understanding of facility conditions, the processes that will take place, the interactions expected or required and to provide open communications of anticipated planning schedules.

The volume of sodium to be measured from systems should be accomplished by calculations of remaining sodium. If the potential exists for negotiation, the regulator should be educated on the relative safety, costs, risks and potential for environmental impacts from the presence of bulk or residual sodium. This activity can significantly reduce the cost to decommission a fast reactor.

A major benefit to the fast reactor community would be a reactor whose cost to decommission was 'equal or near' the cost for a water-cooled reactor (PWR or BWR). Clearly, there will be significant disagreement from both within and outside of the fast reactor community. However, until a common goal is established, this or another one, and is embraced by the fast reactor community, no integrated progress will be made.

Any future fast reactor design should, as a fundamental design criteria, require all sodium systems be provided with an effective sodium removal capability, i.e. draining. In many cases techniques will be simple and inexpensive to implement. In other systems, reactor vessels or steam generators for example techniques will need to be debated.

Current methods for the reaction of residual sodium in situ are being studied, and applied at the EBR-II, e.g. passivation. These techniques should be seriously considered for routine application and any necessary research and development should be pursued. The formalization of specific requirements, suggested applications and process limitations should be developed.

New techniques for the removal of sodium, both bulk and residual, should be pursued for both current reactors as well as implemented into any new fast reactor designs. Techniques should address the entire range of possible alternatives, without limitations of currently existing technologies. Potential future in situ techniques should be identified, prioritized and fully developed allowing the decommissioning of fast reactors to be forthright and cost effective.

4. CONCLUSION

The EBR-II decommissioning activities performed have been discussed. These activities were performed safely, effectively, efficiently and on schedule. Of primary interest are those recommendations stemming from the lessons learned in performing the decommissioning activities. Goals have been suggested for future reactor designs and current decommissioning activities. Finally, the decommissioning experience from the EBR-II has resulted in discussion of relevant Coordinated Research Program topics.

SODIUM REMOVAL & DECONTAMINATION PROCESS AND DECOMMISSIONING CONSIDERATIONS FOR THE PFBR COMPONENTS

M. RAJAN, C.S. SURENDRAN, V. GANESAN, B.S. SODHI, S.C. CHETAL, R.D. KALE, R. INDIRA, A.K. BHADURI
Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, India

Abstract

The Prototype Fast Breeder Reactor (PFBR) of 500 MWe capacity using sodium as a coolant is designed at Indira Gandhi Centre for Atomic Research, Kalpakkam, India. The important components such as sodium pump, intermediate heat exchanger, safety rod drive mechanisms, fuel and other sub-assemblies etc. are to be cleaned and decontaminated before doing any maintenance work or disposal. The sodium removal process for PFBR components envisages reaction of water vapour with sodium in the presence of carbon-dioxide at temperature below 80°C or less. By varying the gas flow rate, the sodium-water reaction rate is controlled. The core sub-assembly is washed with steam and nitrogen mixture to remove sodium sticking on the surface. Detailed studies were carried out and the process was finalized. The choice of chemical decontamination process depends mainly on whether the component is to be reused or to be disposed. Sulphuric acid is planned for decontamination in the presence of phosphoric acid. The choice of hard facing material for different components of PFBR is aimed at keeping induced activity to the minimum for maintenance and decommissioning purposes, and also to reduce the shielding thickness required for the component handling. Based on the investigations, it has been decided to select nickel-base Colmonoy, in place of the cobalt-base Stellite, for hard facing of the NSSS components of PFBR.

1. INTRODUCTION

The Prototype Fast Breeder Reactor (PFBR) is a sodium cooled pool type reactor of 500 MWe capacity. The reactor is being designed at IGCAR, Kalpakkam, India. It is planned to construct the reactor at Kalpakkam. In the PFBR plant Nuclear Steam Supply System (NSSS) components are either wetted with primary sodium or secondary sodium or some time both. Sodium cleaning is to be done before undertaking any maintenance work or disposal. The radioactivity of the primary sodium is significant compared to the secondary sodium. In this paper, primary sodium cleaning and decontamination of the components planned for PFBR are discussed. The choice of hard facing materials for NSSS components are also discussed.

2. SODIUM REMOVAL

During the operation of the reactor there are two types of components, which are to be cleaned from sodium. These are:

- Core components; and
- Primary components.

Figure 1 shows the location of the facility for sodium cleaning in the nuclear island. The core components are handled by fuel handling machines where as the primary components are handled by respective flask. The components that are to be decontaminated are removed from the reactor when the reactor is in shutdown condition. The handling of the core components is based on fixed time duration and interval in the reactor. For the primary components, it is as and when required.

The core component is not meant for decontamination but for reprocessing and the primary component needs to be decontaminated. The cleaning time of the core component is to be minimum and to be completed during the fuel handling campaign, which is 20 days. For the primary component, the cleaning of sodium can be done during reactor operation as well as shutdown condition. These considerations are taken during conceptual design stage of the reactor to evolve better design considering safety, economy, safe-disposal of the components and safe decommissioning of the plant [1 – 3].

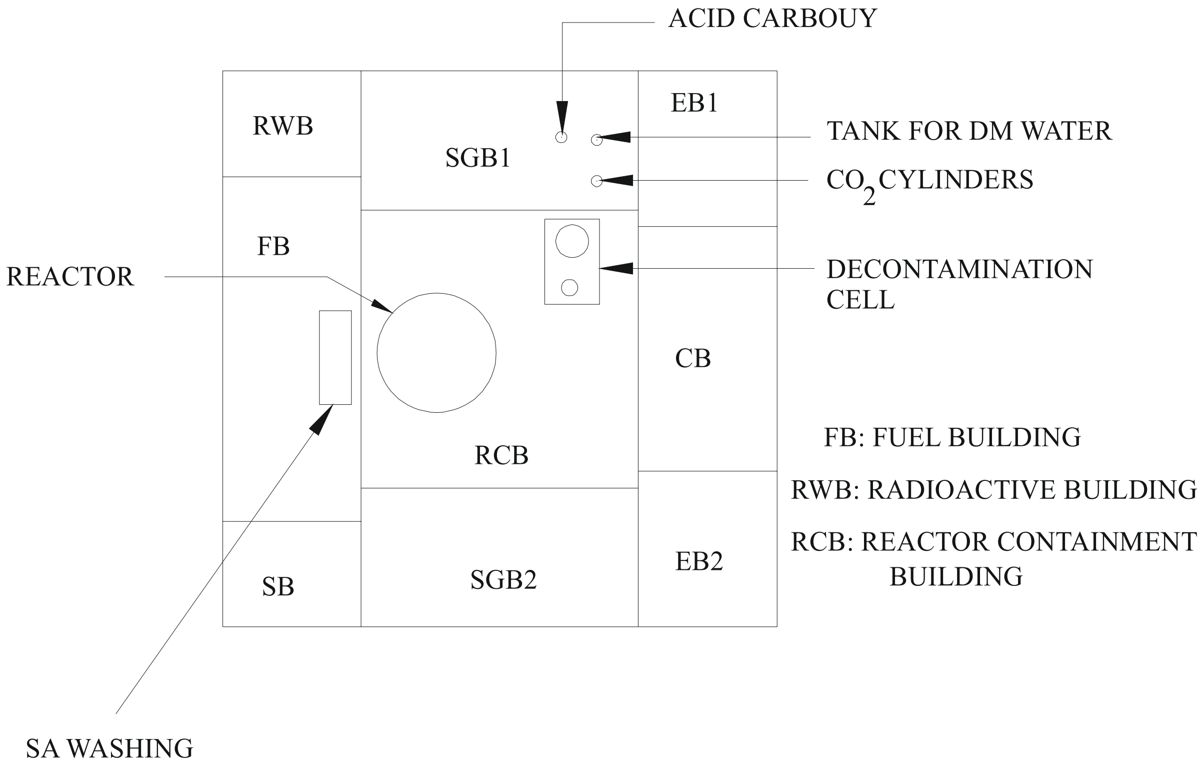


FIG 1. Location for sodium cleaning in nuclear island

The sodium cleaning system has the following design features. A schematic flow sheet of the design features is given in Fig. 2. The main objectives are:

- Minimize sodium sticking on the component;
- Safe operation;
- Minimum quantity of waste generation;
- Fast cleaning;
- Effective cleaning;
- No damage to the component.

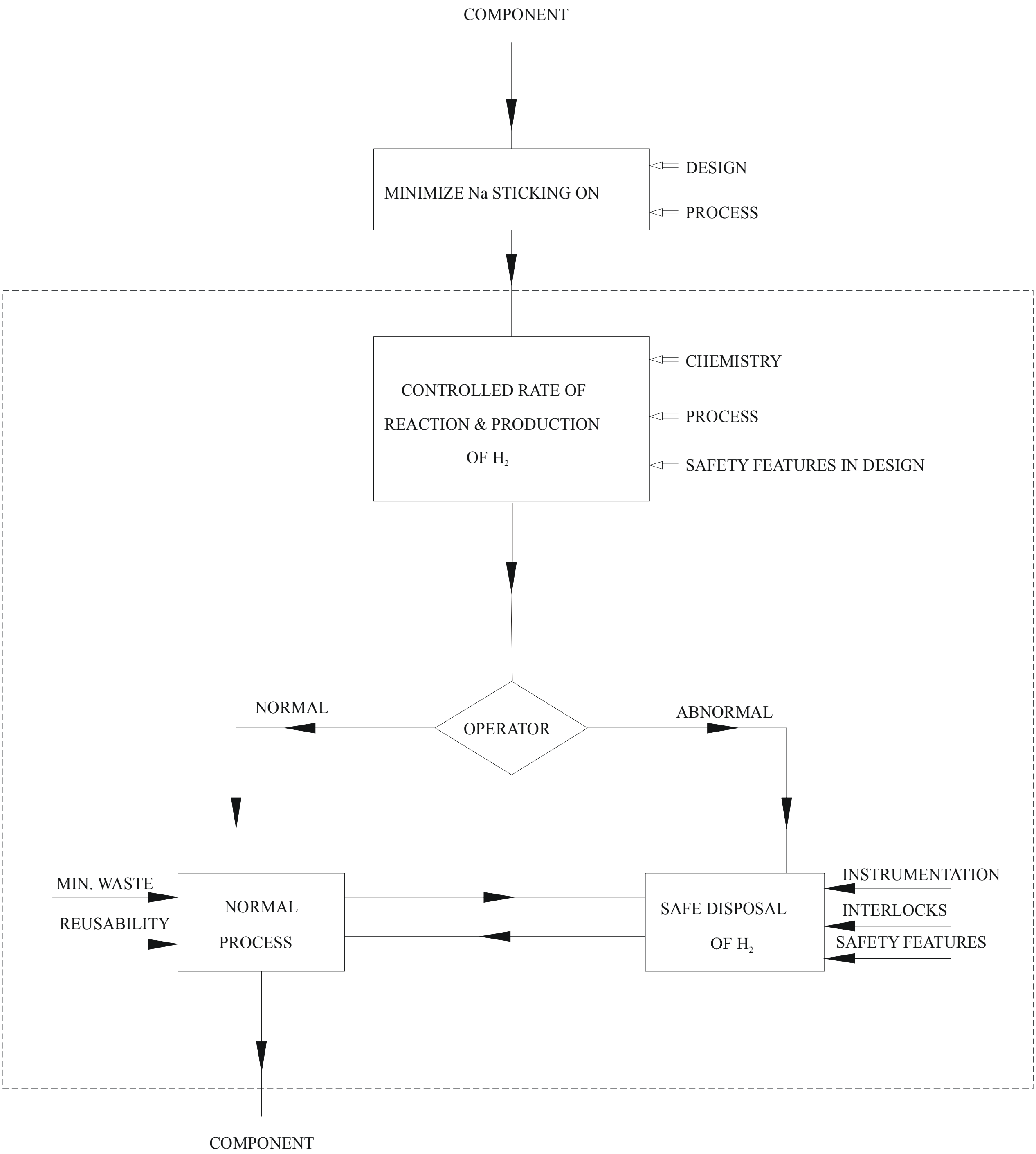


FIG. 2. Shematic flow sheet of the design features.

2.1. Sodium removal from core components

The core components are spent fuel assembly (SA), irradiated blanket SA and irradiated absorber SA. There are 94 core SA to be cleaned from sodium in eight months full power reactor operation. The core SA coming out of the reactor is handled by cell transfer machine. This machine picks up the core SA from the ex-vessel transfer position and keeps it in the washing vessel. The flow sheet of the SA washing is shown in Fig. 3.

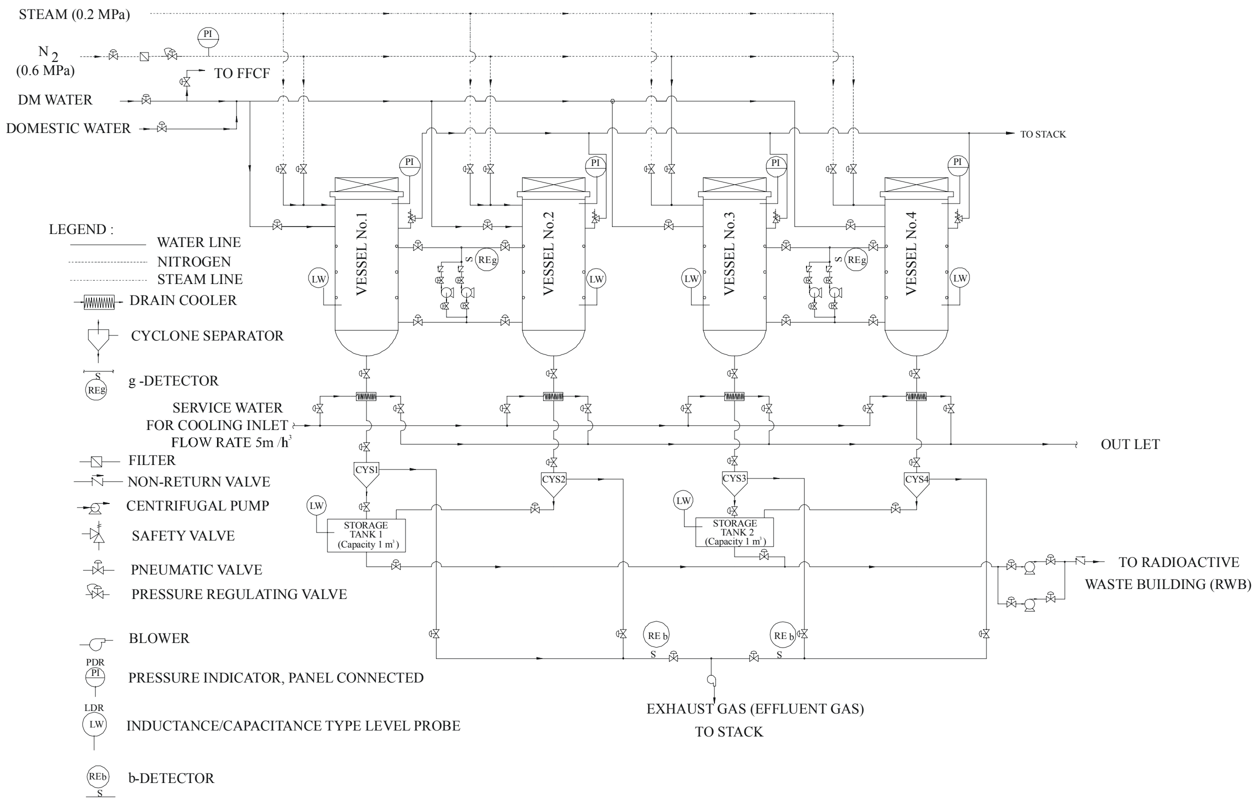


FIG. 3. Flow sheet of the SA washing.

Since the fuel SA is having maximum of 5 kW decay heat, nitrogen at 50 Nm^3/h is required to cool the fuel SA. In order to minimize the sodium sticking on the core SA, the nitrogen flow helps to drain sodium as much as possible at the ex-vessel transfer position while picking the core SA by cell transfer machine. Also, the dripping of sodium in the transfer pot of the Inclined Fuel Transfer Machine (IFTM) is enhanced by holding the SA at an elevated location for a short duration. The time required for discharge of one core SA from the reactor is about 3 hours so that the duration of sodium cleaning of a SA shall be less than 3 hours. It is envisaged that 3 hours duration is required for washing operation. However in the SA washing facility two SAs can be cleaned simultaneously. Also two stand-by vessels are provided at the facility. In washing process, it is possible to identify the breached and non-breached SA. The estimated sodium sticking on a core SA is about 2 kg. Considering the time of washing and cooling requirements, steam cleaning is selected for the sodium cleaning of the core SA. The concentration of hydrogen at the exhaust is monitored and always maintained at less than 2% by varying the steam flow. The steam cleaning is safe and fast, but it lacks the effective cleaning from the crevices. However, water rinsing process is used after steam cleaning. The water is sprayed and re-circulated for about 30 to 40 minute to identify it lacks the effective cleaning from the crevices. However, water rinsing process is used after steam cleaning. The water is sprayed and re-circulated for about 30 to 40 minute to identify the activity level in water using gamma-detector for ensuring whether the SA is cleaned is breached or non-breached. The exhaust gas during steam cleaning is monitored for beta-activity to identify whether the SA is breached. The gas effluent is send to the stack and the liquid effluent is sent to waste treatment. The SA is washed with DM water before transferring it to the storage pool containing water. From the storage pool the fuel SA is taken for reprocessing.

2.2. Sodium removal from primary components

The primary components are classified into:

- Large component; and
- Small component.

The large components are IHX, primary sodium pump, DHX, transfer arm and transfer pot of IFTM. The small components are CSRDM, DSRDM, FFLM, bulk sodium delayed neutron detectors and clad rupture detectors in argon. The large components are handled by one bigger flask and the small components are handled by another small flask. The sodium cleaning facility is also meant for decontamination of the components. This facility is called decontamination facility, which is provided inside RCB. The safety of the plant was analyzed due to the presence of water and potential hydrogen liberated from the facility and was found satisfactory. While removing the component from the reactor by using flask, sufficient time is allowed to drain sodium sticking on the component. For the IHX, the sodium wetted length is 10 m, and while taking it using a flask each 1 m raise of IHX into the flask, one hour time is given to drain sodium from it. This will ensure minimum quantity of sodium sticking on it by avoiding sudden cooling and solidifying of sodium. Thus 10 to 12 hours are required for removing IHX from reactor. Of all the primary components, IHX, is having maximum amount of sodium sticking on its surface. The estimated amount of sodium sticking on the surface of IHX is 75 kg. This component when it is being cleaned gives maximum quantity of hydrogen. The hydrogen liberated during accidental rising of water level in Decontamination Vessel - 1, when IHX is being cleaned, was analyzed. For this analysis, instead of admitting water for bubbling, large quantity of water for rinsing was admitted into the vessel due to operator error was analyzed. The pumping of water into the vessel is limited to 1 cm/minute raise in the vessel. Also the pump can start only when the exhaust valve is kept opened. If the valve fails to open before start of the pump, hydraulic relief pot functions. Also a rupture disc with non-return valve is provided to open into the cell when the hydraulic relief pot does not function. Thus in any case the hydrogen fire is avoided. Moreover the cell air is continuously changed during the process of sodium removal. To avoid water entry into RCB, the water required for sodium cleaning is kept in a small tank of 3 m^3 and a large tank of 10 m^3 capacity outside RCB. By limiting the volume of the tanks, the risk of water flooding in RCB is eliminated. The water from small tank is used for carbon dioxide bubbling process and water from large tank for rinsing the component. CO_2 cylinders are kept outside RCB. The flow sheet of the decontamination facility is shown in Fig. 4.

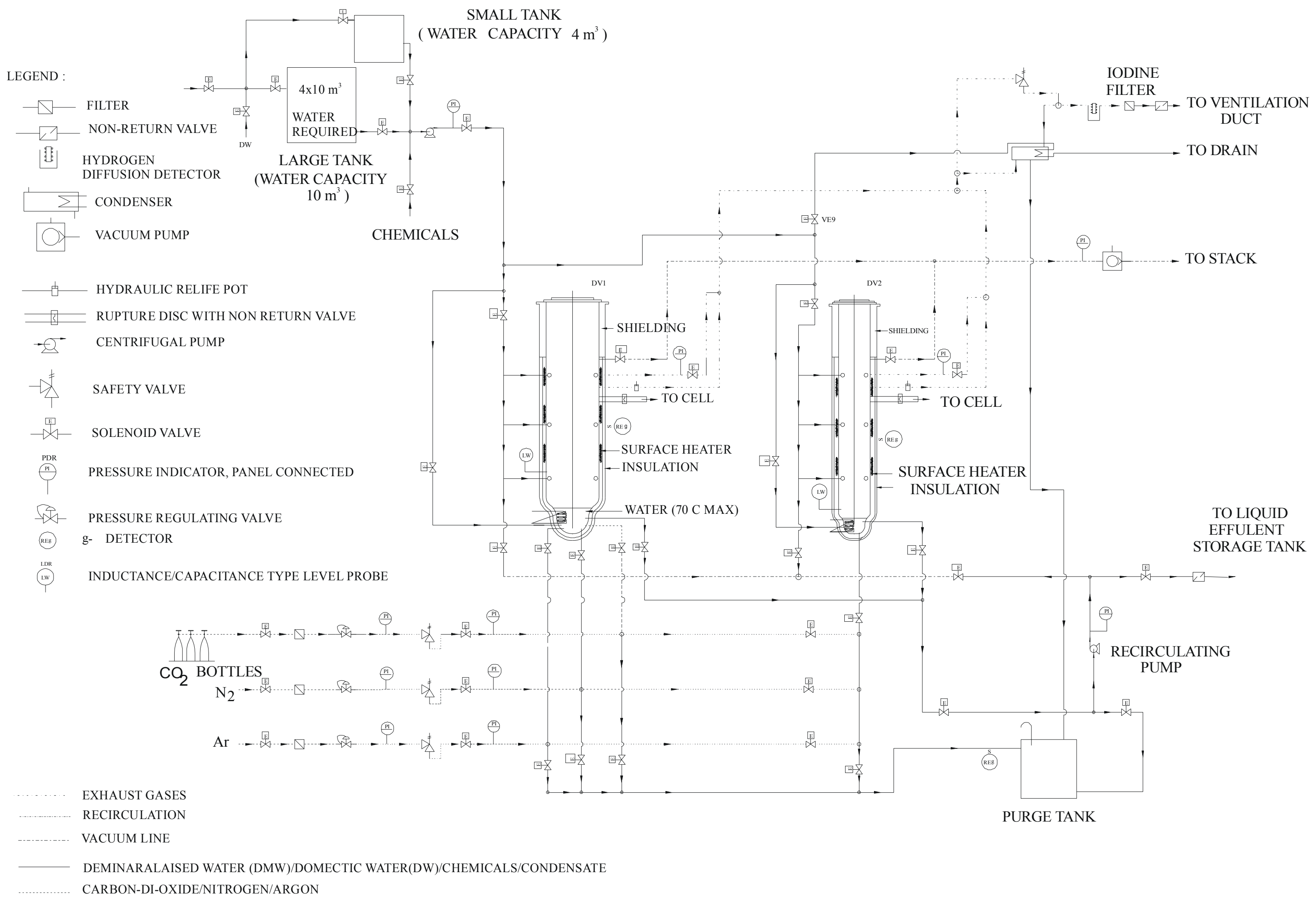


FIG. 4. Flow sheet of the decontamination facility.

In the decontamination facility two vessels are provided. One is for large components and is called Decontamination Vessel - 1 (DV1). The other is for small components and is called Decontamination Vessel - 2 (DV2). The component is brought to the respective vessel and positioned in it. A small quantity of water is admitted to the vessel bottom trough and heated to 70 to 80°C. Carbon dioxide is bubbled through the hot water which will carry the water vapour and react with the sodium sticking on the component. The exhaust gas is monitored and the concentration of hydrogen is always maintained below 2% by controlling the flow rate of carbon-dioxide bubbled.

This method has got better control of sodium water reaction. Also corrosion of the component is not taking place due to absence of alkali. After the bubbling stage, the component is rinsed with water and DM water. The gas effluent is sent to the stack and the liquid effluent is sent to waste treatment. With this the sodium cleaning of the component is over and ready for decontamination.

2.3. Decontamination process

The choice of chemical decontamination process depends mainly on the end use viz. whether the component is for:

- Re-qualification and reuse; or
- Decommissioning and disposal purpose.

Based on the available data on decontamination being followed in other reactor systems and considering the complexity involved in the process, the decontamination of the primary component the following process is used. For the decontamination process sulpho-phosphoric acid process for all component made of AISI 304L, 316L and hard coating is used. The decontamination of the component is carried out in the decontamination facility. After sodium cleaning, the acid (mixture of phosphoric acid and sulphuric acid) for decontamination is admitted to the decontamination vessel to remove surface layer activity. The component is immersed in a solution of 15 g/L of sulphuric acid and 45 g/L of phosphoric acid. The solution at 60°C is circulated for about 6 hours. After this the solution is drained out from the vessel. The component is rinsed with domestic water as well as DM water. Subsequently the water is drained out and the component is dried using surface heaters provided on the surface of the vessel. The component is taken out of the vessel and mopped with cotton before taking it out for doing any maintenance work. Figure 5 shows the flow chart of decontamination process.

Even though mineral acid mixture containing sulphuric and phosphoric acids is found to be a better choice than the organic based solutions for effective decontamination of the components, some future studies are planned for better decontamination solution. However, considering the fact whether the component is for reuse or not and also considering the operation history of the various types of components, it can be stated that one single composition of the decontamination solution cannot be applied to all types of components. Depending upon the requirements, for getting high decontamination factor, the concentration of H_2SO_4 may be increased, or to limit inter granular corrosion concentration of H_2SO_4 is reduced with increase in H_3PO_4. To limit the phosphate release in the waste, concentration of H_3PO_4 may be reduced.

2.4. Radioactive chemical waste generated

During sodium removal from the primary component, liquid chemical in the form of aqueous solution of $NaHCO_3$ and Na_2CO_3 will be generated. The reaction products are a mixture of bicarbonate and carbonate of sodium depending on the availability of sodium hydroxide to react with CO_2 and water. The amount of chemical waste as sodium bicarbonate and sodium carbonate produced during sodium cleaning by CO_2-water vapour process depends on the relative amount of sodium in the component and the availability of carbonate and water vapour. From laboratory experiments, it is found that the layer of $NaHCO_3/Na_2CO_3$ is highly porous allowing the entire sodium sticking on the surface to react, leaving no unreacted sodium on the component. At the end of sodium dissolution step, water is sprayed on to the component. The water in the cleaning vessel is raised to required level in steps so as to immerse the component and circulate to ensure complete dissolution of the reaction products. The contaminated water at this stage is drained. Domestic water is used for second stage cleaning to reduce alkali content in the vessel to a minimum and drained out. Third stage is the chemical solution for decontamination. Fourth and fifth stages involve domestic water and DM water cleaning. The typical liquid waste generated during cleaning and decontamination of primary sodium pump is nearly 260 m^3, which is the maximum expected waste for a primary component. During SA washing about 725 L of liquid waste is generated. The liquid waste is sent to radioactive waste building for treatment and disposal. The gaseous waste generated is directly sent to the stack. The solid waste in the form of waste cotton is covered in polythene bag and sent for disposal.

2.5. Choice of hard-facing materials for NSSS components

The choice of hard facing material for different components of PFBR is aimed at keeping induced activity to the minimum for maintenance and decommissioning purposes, and also to reduce the shielding thickness required for the component handling, which in turn would reduce the flask weight, size of handling crane and loads on civil structures. Induced activity, dose-rate and shielding computations were carried out for the various NSSS components of PFBR hard faced using the same amount of stellite and colmonoy with their actual geometrical configuration and operating conditions.

Hard-facing of NSSS components with cobalt based alloys of stellite, results in difficulties in maintenance, decommissioning and handling, due to induced activity of cobalt-60. Hence Nickel based hard facing materials, colmonoy, and triballoy were considered as alternate choices. Induced activity computations were carried out considering the same amount of stellite, colmonoy-6 and triballoy-700. The activity and dose rate computations have been carried out for 0.25% Co in colmonoy and T700 (same as in the case of austenitic stainless steels).

For the in-vessel components, control and safety rod drive mechanism (CSRDM), Diverse safety rod drive mechanism (DSRDM), failed fuel location module (FFLM) and primary sodium pump (PSP), saturation activity was calculated, as these components are expected to be irradiated for 20 years. A cooling time of 2 or 5 years was considered in the case of grid plate components, as these are expected to be handled only for decommissioning purposes. In the case of CSRDM, DSRDM, FFLM and PSP, a cooling time of 2 days has been considered. For the Control and Safety Rod (CSR) and Diverse Safety Rod (DSR), an irradiation time of two years and cooling time of 2 days was considered [4].

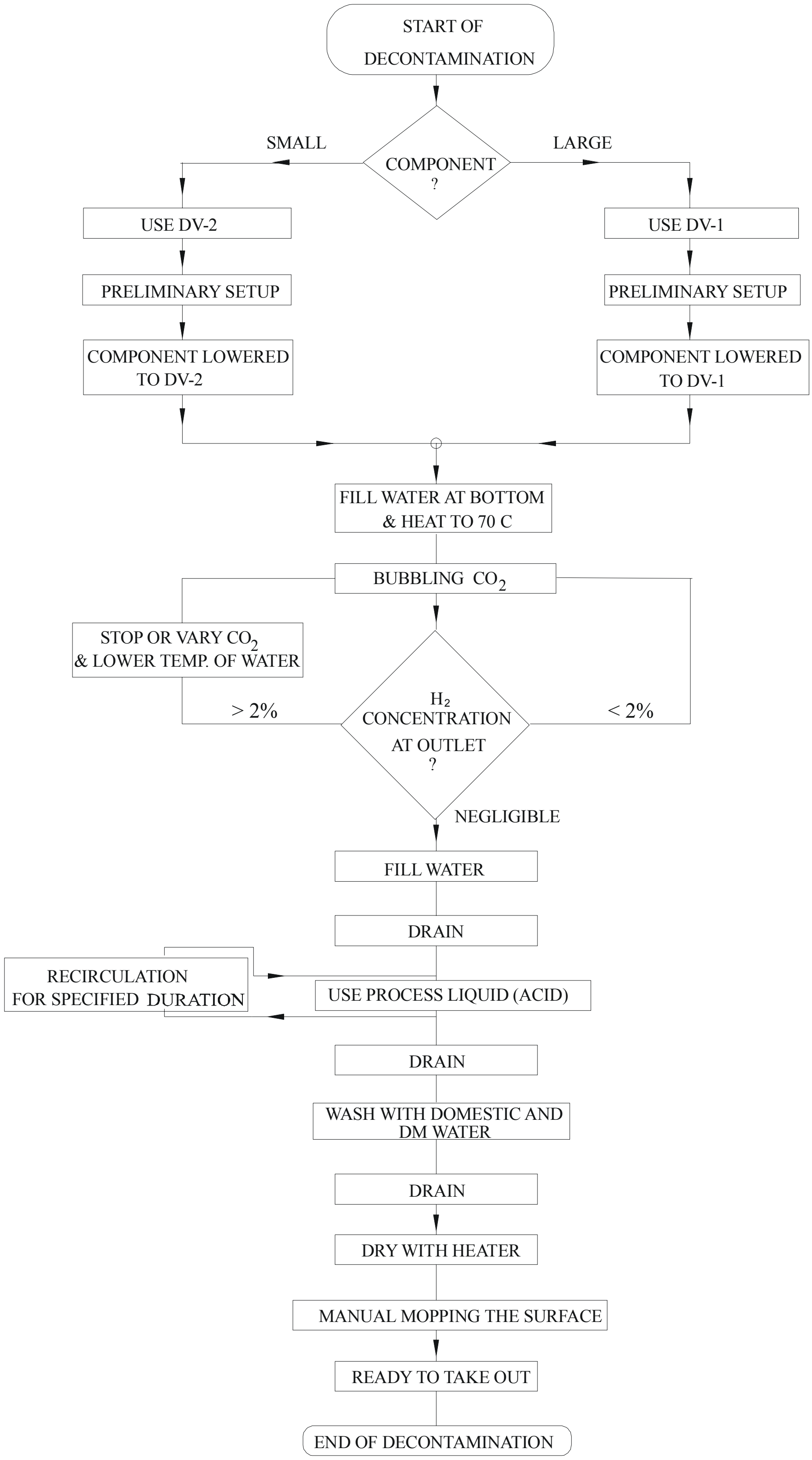


FIG. 5. Flow chart of the decontamination process.

It is found that from induced activity and dose rate considerations, stellite can be replaced by either Colmonoy-6 or Tribaloy-700. Based on fast reactor experience with colmonoy, colmonoy has been chosen. An approximate estimate of the shield requirement assuming a permissible dose rate of 100 mR/h have been worked out. In the following discussions, as the basic SS component itself becomes radioactive and requires shielding, reduction in shield thickness is with respect to the minimum shielding required for the SS component [5].

At the grid plate top sleeve only decommisioning need be considered, since there is no maintenance. It is seen that by replacing stellite the man-rem requirement can be reduced by a factor of 250, while handling grid plate. By replacing stellite in the case of anti-rotation plugs in grid plate, man rem requirement can be reduced by a factor of 30 during decommissioning. For the IHX seal flanges in inner vessel, the man-rem requirement comes down by a factor of 370 during decommissioning, if stellite is replaced.

In the case of CSRDM and DSRDM, by replacing stellite, the lead shield requirement for maintenance of DSRDM and CSRDM can be reduced by 6 and 3 cm, respectively. From the point of view of shielding for IFTM (Inclined Fuel Transfer Machine) and flask for handling, replacing stellite by colmonoy in CSR is highly beneficial. Based on the above considerations, colmonoy has been chosen as hardfacing material for the grid plate components, CSRDM, DSRDM and CSR. For grid plate components, Cr-N coating is also being considered as an alternate option.

REFERENCES

[1] JAMBUNATHAN, D., RAO, M.S., KRISHNAMACHARI, V.S., KASIVISWANATHAN, K.V., RAJAN, M., Experience of sodium removal from FBTR components in its operating phase, paper presented in the Technical Committee Meeting on Sodium Removal and Disposal from LMFRs in Normal Operation and in the Framework of Decommissioning, Aix-en-Provence, France, 1997.

[2] RAJAN, K.K., GURUMOORTHY, K., RAJAN, M., KALE, R.D., Sodium cleaning and disposal methods in experimental facilities, paper presented in the Technical Committee Meeting on Sodium Removal and Disposal from LMFRs in Normal Operation and in the Framework of Decommissioning, Aix-en-Provence, France, 1997.

[3] MASSE, F., RODRIGUEZ, G., Cleaning and Decontamination: Experimental Feed Back from PHÉNIX, paper presented in the Technical Committee Meeting on Sodium Removal and Decontamination from LMFRs, in Normal Operation and in the Frame Work of Decommissioning, Aix-en Provence, France, 1997.

[4] INDIRA, R., Choice of Hardfacing Materials for LMFBR Components, Proc. National Symposium on Radiation Physics (NSRP-13), Mangalore, 1999, Indian Society for Radiation Physics, Mangalore University, Mangalore, India (1999) p. 459.

[5] BHADURI, A.K., INDIRA, R., et. al., Recommended Hard facing Materials for NSSS Components of PFBR, Task Force Report No. IGC/HTF/04, March 2000.

OVERVIEW OF TACIS PROJECT K1.01/98A 'ASSISTANCE TO THE ELABORATION OF A DECOMMISSIONING PLAN' FOR AKTAU BN3-50 NPP (KAZAKHSTAN)

R. BRACHI
Electricité de France/CIDEN (EDF/CIDEN), Villeurbanne Cedex, France

Abstract

This project is part of the On Site Assistance provided by a consortium established between French EDF and Italian Sogin to the Aktau BN-350 NPP, financed by the 1998 budget of Tacis programme of the European Commission (EC). The UKAEA took part in the project that lasted 18 months, since May 2000. This report gives an overview of the outputs of the project: technical studies, study tours and equipment, as well as of interfaces with other international assistance projects to BN-350 NPP decommissioning. Nine technical tasks, addressing the essential topics of nuclear decommissioning, were implemented, aiming to contribute to the elaboration of the decommissioning plan. Topics covered range from plant data collection and regulatory framework analysis to dismantling options studies, including waste management strategy, risk and environmental impact analysis, cost assessment and quality assurance issues.

1. INTRODUCTION

This project is part of the On Site Assistance provided to the Aktau BN-350 NPP, financed by the 1998 budget of Tacis programme of the European Commission (EC). Its implementation was entrusted by the EC to a consortium established between EDF and Sogin. EDF acts as consortium leader trough its engineering unit in charge of nuclear facilities decommissioning (CIDEN). Sogin is the Italian public company that owns, operates and has to decommission four Italian shut-down NPP formerly owned by ENEL.

The UKAEA had agreed to take part in the project that started in May 2000 for 18 months. The terms of reference of the project are stated in the service contract Tacis 00.0061, awarded by EC to EDF/Sogin Consortium in May 2000. The project was co-ordinated by an EDF expert who was hosted, for the project duration, by the Kazakhstan Nuclear Safety Center (NTSC) in Almaty.

This report gives an overview of the deliverables due by the Consortium as per the above-mentioned service contract with the EC. Deliverables include administrative reports (not detailed in this document), technical documentation, study tours and equipment.

Project’s deliverables belongs to the EC and are given to the project Beneficiary, the Mangyshlak Atomic Energy Complex (MAEC) who is the owner operator of the Aktau BN-350 NPP.

Implementation of the project, which covers 9 technical tasks addressing the essential topics of nuclear decommissioning, was to be approached in two main phases: first data collection, then contribution to the elaboration of the decommissioning plan.

1.1. International structure of support to BN-350 decommissioning

In the period that preceded and followed the decision to shut-down the plant in April 1999, Kazakhstan authorities concluded two agreements, first with the US-DOE for it to assist in the elaboration of BN-350 whole decommissioning plan by providing both financing under an ISTC programme and technical review, second with Russian institutes for actually elaborating the decommissioning plan under commercial conditions with ISTC and Kazakhstan funding.

In order to preclude duplication of efforts, Tacis project's terms of reference commit the Consortium to fit in this pre-existing working organisation.

Two ways could be envisaged: either providing direct technical assistance to the Russian institutes, or making available European experience and know-how in the form of technical reports that would be handed to the Beneficiary, then transferred by it or rather by NTSC to the Russian institutes. As mentioned in administrative reports to the EC, the first approach could not be worked out and the Consortium therefore settled on the second alternative.

As a result the Consortium's work had to focus on providing the Beneficiary with technical advice, options weighing, and qualitative judgements, instead of describing the decommissioning tasks themselves. The resulting reports that have been or are being issued for each one of the nine technical tasks are mainly intended to be used, concurrently with other sources of information and knowledge, by the Russian institutes to establish the documents of the decommissioning plan, although in cases when these institutes could not improve on them, they might be included into the preliminary decommissioning plan without modifications.

The first draft of the complete BN-350 NPP decommissioning plan was set up by the NTSC end of January 2002 and will be presented by to a co-ordination expert meeting convened under the supervision of the IAEA in Vienna from 19 to 21 March 2002.

1.2. Data collection issue

As stated in the terms of reference of the project, "no dismantling plan can be devised, let alone carried out, without a complete knowledge of the plant, site, regulation, environment, industrial and organisational details which are the practical circumstances of and the reference for its implementation".

Since the draft of the 'Terms of reference' had been approved by the Beneficiary earlier, it had been implied by the Consortium that data would be forthcoming during the initial phase of the project, even if more or less organised. Unfortunately, mainly due to the poor document's archiving on the site, the actual implementation of the project was most of all characterised by the dearth of technical information. Room by room data collection, was organised by the Consortium and entrusted to a local sub-contractor that produced a rough global assessment of materials with figures of their masses, volumes, natures and radiological contents.

Outputs from this preliminary survey will be later on improved and completed in the frame of NPP radiological characterisation survey[1] to be implemented during the plant preparation to safe enclosure.

One special mention yet should be made of the regulatory framework, where a quite complete set of the documents in force in Kazakhstan was gathered, copied and provided by NTSC in CD-ROM format. This however occurred 16 months after project initialisation.

This situation was not particular to Tacis' part of the project, and we heard similar complaints from our American colleagues. Russian contributors (mainly VINIPIET) were in a particular

[1] Activity also supported by another Tacis project (K1.01/98 B), aimed to deliver equipment and methodology for radiological survey.

situation since they owned the plant design documentation. Discussions about their providing some of this information to the Consortium failed to produce results.

1.3. Documentation Exchange System (DXS)

This system is provided to NTSC, as directed by the Beneficiary. It comprises a main server located in Almaty running the dedicated software, and distant PCs in Aktau. The internet connects the server to the distant partners which, besides Aktau, include KAEC, KATEP, the IAEA, the Consortium and other partners agreed by Kazakhstan, such as Russian institutes, ANL, Japanese companies.

The aim of the system is to create a reliable archive, using duplicated CD-ROMs, of documentation to be used for BN-350 decommissioning and to enable distant users to access it. The supply of this system and of the know-how transfer provided to NTSC and BN-350 experts, come in excess to the assistance initially foreseen in the Tacis contract.

2. THE TECHNICAL REPORTS PROVIDED BY TACIS PROJECT

Nine technical reports were issued by the Consortium in the framework of this project. Their content and present status is described as follows.

2.1. Data gathering, tools for elaboration and reporting

This report contains a brief description of the plant archives, the complete set of technical, financial and environment data gathered to this day and used for the elaboration of the other technical reports of the project, the justification and presentation of the specifications for a documentation archive and exchange system (DXS), a brief report on the commissioning of the central part of this system (the server located in the premises of NTSC) and the specifications for the Aktau station. The description of the archives and set of plant and financial data exist as a draft written by Sogin with help from a local subcontractor. The DXS part was written by NTSC (report on commissioning will be issued after this is completed by end of first quarter 2002). The specifications for the Aktau station have been written by Sogin in November 2001. Purchase of equipment for the Aktau branch of DXS is in progress, in accordance with Tacis procurement procedures.

2.2. Chain of responsibility and management

This task was cancelled at the initiative of the Beneficiary. A short report will be issued by EDF with diagrams of the main Kazakhstan organisations, with the exception of MAEC. These diagrams can presently be found in the project organisation report.

2.3. Environment data

A small quantity of data on the state of environment on and around the plant site (Caspian sea, soil, underground water, winds, population, economy, radioactive pollution, pollution by heavy metals) were communicated by Kazakhstan organisations, and part of this only orally, in informal meetings with experts. This information will be entered into the data gathering report (see Section 2.1) with every qualification as to its accuracy.

Beside, in addition to the planned outputs of the project, the EC agreed to support the publication of a book on Kazakhstan radiological pollution, based on date provided by a former study financed by ISTC. Publication is managed by NTSC.

2.4. Waste facilities

As it turned out from discussions with Kazakhstan experts, besides temporary storage areas and buildings on plant site, possibly not suitable for further use, no facility is available in Kazakhstan to dispose of radioactive waste. A study is underway by KATEP, on orders from the government, to study options for the disposal of radioactive waste in the Aktau area. The bulk of the waste disposed of in this facility would come from closed uranium mines, but some might be accepted originating from BN-350, provided an impact study shows they are compatible with the characteristics of the chosen site. This study is reported on by EDF in the first part of the technical report on the disposal pathways for BN-350 dismantling waste. The same document makes proposals for the temporary safe storage of conditioned radwaste on plant site. The document exists as a final draft, pending comments from Kazakhstan organisations.

2.5. Financial data

Financial data obtained during the course of the project (staff projections, salaries, water and power consumption) are included in the report on plant data (see Section 2.1). When the study of decommissioning options involved a judgement on their comparative costs, this judgement was made based on the European experts' knowledge of similar projects in the CIS or Eastern Europe, when available, but not on local data. The financial data and some assumptions used for the assessment of decommissioning costs were included in the costing document issued by Sogin in December 2001.

2.6. Rules and regulations

A report on existing regulations as been issued as a draft by NTSC, with a conclusion pointing to some shortcomings. A complete set of regulations governing decommissioning-related activities was issued by the NTSC as a CD-ROM comprised of scanned images of the texts in Russian language and a summary with automatic links to the texts. This CD-ROM will be re-issued in English later on.

Comments from EDF on required complements to the regulations can be found in the reports on Operational Waste and Decommissioning Waste for issues connected to these topics. Since the Consortium was informed by NTSC that a contract had been agreed with Argonne National Laboratory (USA) for the latter to provide advice on required complements to Kazakhstan regulation, further foreseen developments on this issue were dropped to avoid duplication.

A limited support to complement regulations pertaining to waste management strategy, will be provided, during 2002, by a subcontract awarded by the Tacis Consortium to NTSC (under negotiation).

2.7. Quality assurance

Task’s completion report is under elaboration by a local Consortium’s consultant. It will include three mission reports by the Consortium QA expert to Aktau, a small number of documents (programmatic declaration, procedures) produced by the Aktau plant management as a result of these missions, certificates of training of Aktau QA experts by the Bureau Veritas in Moscow, and comments on the national Quality Programme by French QA experts.

2.8. Waste management

A first short report by EDF covers Operational Waste and, for respectively solid, liquid radwaste and sealed radioactive sources, provides an analysis of the waste, a judgement about regulatory issues and technical proposals for processing the waste[2].

A second report by EDF covers Decommissioning Waste and, based on plant data and the international experience of managing waste from dismantling nuclear plants that are extensively described, mentions technical processes to address BN-350 various categories of radioactive and hazardous waste. It provides information about efforts directed to the creation of a national radwaste repository that, in addition to waste resulting from closed industrial facilities, might take up part of BN-350 radwaste. With the assumption that such a repository might not be ready in time for SAFSTOR preparation, it proposes temporary waste storage facilities to be arranged in some BN-350 existing buildings.

The third report by Sogin covers Conventional Waste. It provides general directions as to how each type of waste should be handled.

Further elaboration of waste management strategy, will be supported, during 2002, by a subcontract awarded by the Tacis Consortium to NTSC (under negotiation).

2.9. Risk and environmental studies

Task's completion report will be issued by end of first quarter 2002. It will include 3 studies produced by NNC with results of calculations that assess the consequences of hypothetical events occurring while BN-350 is being decommissioned, in terms of atmospheric releases, propagation of contamination in the ground, and human mortality and morbidity, and a 4th study, also by NNC, presenting a general risk analysis of the main decommissioning phases. An appendix relates to the provision of codes and training that enabled NNC to carry out these studies.

Kazakh expert training to use of the above-mentioned codes is being completed during these days (Sogin expert mission to Kurchatov NNC premises, end of February 2002).

2.10. Dismantling scenario

Task's report presents, based on a plant description making use of available plant data (see Section 2.1), a systematic analysis and a reasoned, weighted comparison of elementary options, essentially focussing on the definition of the SAFSTOR, or 50 years confinement, state of the facility which is the strategy chosen by the government of Kazakhstan. As a result two preferred decommissioning options are identified and described, the first aimed to overall optimisation, but resulting in substantial expenditures in the initial 15 years period, the second intended to minimise initial expenditures, while admitting sizeable care and maintenance expenses during the safe enclosure period. Required ancillary facilities are listed for both options.

This report by UKAEA and AEA-Technology has been issued as a final draft, pending comments from Kazakhstan organisations.

---

[2] Activity also supported by another Tacis project (K1.01/98 C), aimed to deliver equipment for partial treatment of operational LRW stored on the BN350 NPP site.

2.11. Financial evaluation

The reports presents an overall costing of the main decommissioning phases (Safe enclosure preparation, Care & Maintenance, Final Dismantling) for each essential expense count (such as staff, waste, care & maintenance, etc.) and for both preferred options identified in Section 2.10. The costing is based on overall assessments, since detailed data which would have allowed to assess the elementary spending for each operation are not available. Fuel and sodium processing are not included in the costing, as the associate tasks are mainly a matter of the assistance programme of US-ANL, which agreed to provide the corresponding figures. Task's completion report was issued by SOGIN in December 2001.

3. STUDY TOURS

In association with the implementation of the project tasks, specific study tours in Europe were organised by the Consortium. Among them, the following ones worth to be mentioned:

- Study tour to the PFR at Dounreay, in Scotland for BN-350 experts involved in waste management and contractors management;
- Study tour in France to NPPs under decommissioning (Brennilis, Creys-Malville) and waste storage site (site de l'Aube);
- Study tour in Italy to NPPs under decommissioning (Garigliano, Latina) and to Sogin headquarters meeting experts on decommissioning cost assessment and staff management.

Consortium experts, in charge of each project's task, executed specific missions to the BN-350 site. Project review meetings were organised in Kazakhstan and in Europe. The Kazakhstan Atomic Energy Committee was associated to these reviews, in consideration of its position of Nuclear Safety Authority.

4. PROVIDED EQUIPMENT

4.1. In Aktau

The specifications for the DXS local station were established by agreement between the local representative of the consortium and the Beneficiary. In principle it is made up of 3 PCs. These will be bought locally, by the Consortium's local representative, under Tacis procurement procedure, in progress at date. Additional documentation archiving and editing equipment (PCs, scanners, printers & so) will also be provided to the plant in order to improve the documents storage and management conditions.

4.2. In Almaty

The DXS server (see Section 1.3) supply contract is near to completion. The server will be commissioned in March 2002.

5. CONCLUSION ON PROJECT'S DELIVERABLES

The deliverables provided through the K1.01/98A project comply with the terms of reference of the project. All the technical documents were mailed or handed to it as drafts in English and in Russian at various stages. Only few comments from the Beneficiary have been received to this day. This is not very surprising, given that the actual application of the

European know-how transfer is to nourish the official decommissioning plan entrusted by the Kazakhstan side to the Russian institutes.

The review of the plan during the spring of 2002, requested from the IAEA by the Kazakhstan government, is likely to highlight the contribution of Tacis.

The next step of development of the plan, which the Kazakhstan side calls the 'decommissioning project', will undoubtedly build on the deliverables of the present Tacis project, as well as on the comments and questions they elicited from the experts commissioned by the IAEA to review the decommissioning plan.

6. GENERAL CONCLUSION

The EDF/Sogin Consortium believes that the assistance provided through K1.01/98A project complies with the Tacis project TOR as a whole. The achievements of each project task will be less or more valuable for the elaboration of the decommissioning plan depending, among others, on the relevance given by the Kazakhstan side on various items of the plan.

The international peer review of the plan, envisaged under the supervision of the IAEA will help the Kazakhstan side to take the best profit of Tacis contribution.

The Consortium believes that several aspects of the plan need a further development and possibly further contribution of western expertise. Among them the Consortium highlights the issues related to the characterisation, sorting, conditioning and storage of solid waste either issued from operation or expected to be produced during decommissioning implementation.

The decommissioning project implementation will require huge funding and Kazakhstan is open to accept foreign contribution and technical assistance.

The USA are already committed to support relevant and urgent activities, designed by the decommissioning plan for the safe enclosure phase preparation, as spent fuel and sodium safe disposal.

Japan Company RANDEC is also committed to provide technical assistance, namely for evaluation of radioactive inventory and workers exposure, waste conditioning and project management.

The EDF/Sogin Consortium is preparing, in agreement with Kazakh partners, a new proposal to be submitted to EC, aimed to provide further support activities to be implemented to convert the BN-350 NPP to safe enclosure.

GENERAL REVIEW OF THE DECOMMISSIONING OF LIQUID METAL FAST REACTORS (LMFRs) IN FRANCE

G. RODRIGUEZ
Commissariat a l'Energie Atomique (CEA), Saint-Paul-Lez Durance Cedex, France

R. FRITH
Electricité de France/CIDEN (EDF/CIDEN), Villeurbanne Cedex, France

M. BERTE
Framatome-ANP/Novatom, Lyon Cedex, France

Abstract

This paper gives a general review of the decommissioning of Liquid Metal Fast Reactors (LMFRs) held since the last decades in France. It summarizes the main steps to process the decommissioning of RAPSODIE and SUPERPHÉNIX. This experimental feedback can highlight several recommendations and a general approach for the decommissioning of future LMFRs still in operation.

1. INTRODUCTION

In the 60's and mid 70's, nuclear power was expected to develop very quickly across the world. The Fast breeder reactors were then developed in every nuclear countries with merely the same concept: liquid metal cooled (sodium or potassium sodium alloy) and mixed (U, Pu) oxide fuelled. Thus all countries have developed a nuclear program based on the building of research breeder reactors (RAPSODIE, KNKII, DFR), than there was the building of some prototype reactors (PHÉNIX, PFR, BN-350) and at least the design or building of some commercial reactors (SUPERPHÉNIX, BN-600, EFR (project)). As it is clearly expressed in paper and mentioned in Ref. [1]: "Seen from the early 70's, Fast Breeder Reactors were second generation plants, whose turn should only come when thermal reactors will have significantly depleted the ^{235}U resources. On the other hand, they offered the prospects of unlimited energy supply". In the mid of the 80's it appears that the demand of nuclear energy was less important and crucial than expected and in all European countries the development program for breeders reactors continuously decreased. It has been completely stopped in several countries such as Germany, Italy and latter in the United Kingdom. This new strategy was accompanied by the definitive shutdown of several breeder reactors. Thus the question of the decommissioning of several LMFRs in Europe and even in the rest of the world became a major subject. The interest in the decommissioning of LMFRs raised with the final shutdown of prototype reactors such as PFR and the prematurely shut down of the Superphénix reactor in summer 1997. Table 1 gives a general historic of all the LMFRs developed in the world and their situation at the present time [2 – 5].

It can clearly be seen that a majority of reactor built in the western countries are now stopped and there is therefore in all these countries (USA, Western Europe, and ex-USSR) development of studies and operations for the decommissioning of LMFRs. Thus the objective of this review is to present the different steps of the decommissioning of French LMFRs (RAPSODIE and SUPERPHÉNIX) in order to present the particularities of these decommissioning techniques and to highlight the state of the art in decommissioning Liquid Metal Fast Reactors, that can be useful when time will come for the reactor still in operation to be stopped. Even for fast reactors in project, it happens that taking into consideration the future decommissioning of these reactors in the design stage will become now an increasingly demand of safety authorities to accept the project for its future realization.

TABLE 1. SUMMARY LIST OF LMFR POWER REACTORS, PLANTS AND PROJECTS

Country	Name	Thermal Power (MWth)	Electric Power (MWe)	Coolant	Type	Date of first criticality	State
USA	Clementine	0.025	0	Hg	ER	1946	FS: 1952
USA	EBR 1	1.4	0.2	NaK	ER	1955	FS: 1963
USA	LAMPRE	1	0	Na	ER	1961	FS: 1965
USA	EBR 2	60	18	Na	ER	1956	FS: 1993
USA	FERMI	300	100	Na		1963	FS: 1972
USA	SEFOR	20	0	Na	ER	1969	FS: 1972
USA	FFTF	400	0	Na		1980	S: 1992
USA	Clinch River	975	380	Na	DR	X	CS
Ex USSR	BR 1	0.03	0	Na	ER	1955	FS:
Ex USSR	BR 2	0.2	0	Hg	ER	1956	FS: 1958
Ex USSR	BR 5-BR 10	5-10	0	Na	ER	1958/73	IO
Ex USSR	BOR 60	60	0	Na	ER	1969	IO
Ex USSR	BN 350	1000	350	Na	DR	1972	FS: 1999
Ex USSR	BN 600	1430	600	Na	PR	1980	IO
Ex USSR	BN 800	1970	800	Na	DR	?	P
France	RAPSODIE	24-40	0	Na	ER	1967/70	FS: 1983
France	PHÉNIX	560	250	Na	DR	1973	IO
France	SUPERPHÉNIX	3000	1200	Na	PR	1985	FS: 1998
France	SPX2	3600	1500	Na	PR	X	AP: see EFR
GB	DFR	72	15	NaK	ER	1959	FS: 1977
GB	PFR	600	250	Na	DR	1974	FS: 1994
GB	CDFR	3300	1320	Na	PR	X	AP: see EFR
Germany	KNK 1-KNK 2	60	20	Na	ER	1972/77	FS: 1991
Germany	SNR 300	730	300	Na	DR	X	CS
Germany	SNR2	3600	1500	Na	PR	X	AP see EFR
Japan	JOYO (JEFR)	100	0	Na	ER	1977	IO
Japan	MONJU (JPFR)	714	250	Na	DR	?	IO
Japan	DFBR	2000	800	Na	PR	?	P
India	FBTR	40	15	Na	PR	1985	IO
India	PFBR	1210	500	Na	DR	?	P
Italy	PEC	140	0	Na	ER	X	CS
China	CEFR	65	23	Na	ER	?	P
Rep. of Korea	KALIMER	392	150	Na	DR	?	P
EEC	EFR	3600	1500	Na	PR	?	AP*

ER: Experimental Reactor
DR: Demonstration Reactor
PR: Powered Reactor
FS: Final Shutdown
S: Stopped
IO: In operation
P: Project
AP: Abandoned Project
CS: Construction stopped
* The project was completely achieved and then abandoned.

2. SPECIFICITIES IN THE DECOMMISSIONING OF LMFR

For decommissioning operations, the main specificity in LMFRs is the nature of the coolant that is, for all the reactors ever built, always a liquid metal and that has to be considered as a chemical waste at the final shutdown of the reactor. In its form, the liquid coolant (sodium or sodium-potassium alloy) cannot be considered as a stable nuclear waste due to its chemical properties: strong reaction with water and potential ignition with air when liquid [6, 7]. Thus in the decommissioning phases it will be necessary to consider the transformation of this coolant into a stable chemical product.

The treatment of sodium or NaK will be encountered at every step of the decommissioning operations:

- During the defuelling phase, the assembly will be covered by a residual film of sodium (or NaK) that has to be removed before storing the elements in the pool;
- Every component extracted from the reactor will also be covered by a film of sodium and can sometimes retain residual amount of sodium. This sodium will have to be removed prior to the dismantling of the components;
- The metallic coolant coming from the primary and secondary circuits will have to be chemically treated in order to transform these large amounts of metallic radioactive products (several tons to several hundred of tons) to stable products;
- The primary vessel drained from the primary coolant and the secondary circuit drained from the secondary coolant will have inside some residual amount of liquid metal stuck to the surface or retained inside the structures as non-drainable retentions. Hence, this particular situation will have to be solved prior to the study of the decommissioning of the main structures;
- At least, the decommissioning of a LMFR will produce several secondary wastes full of sodium and highly activated or contaminated i.e. cold traps or cesium traps, etc… A specific treatment for these wastes will have to be found.

Thus in the field of the decommissioning of LMFR the term of 'sodium waste treatment' is generally used. This term gathers all the processes that can be used to transform sodium to a chemical stable product to achieve the global decommission of the reactor [8].

These specific techniques applied for the decommissioning of LMFRs and in particular, the specific processes developed at every step for sodium treatment will be presented in the description of the decommissioning operations of the following reactors: RAPSODIE and SUPERPHÉNIX in the Sections 3.1. and 3.2, respectively.

3. DECOMMISSIONING OF FRENCH FAST REACTORS [9]

3.1. RAPSODIE reactor

3.1.1. History and general description of the reactor [2–4, 10]

RAPSODIE reactor is located in South of France, at the Cadarache CEA Centre (Figs. 1 and 2). The first studies of RAPSODIE were in 1958. Start of construction was in 1962 within an association of CEA and EURATOM. The reactor was of the loop type and had two cooling circuits of 12 MWth heat capacity removal each. First criticality was reached on 28th January 1967. The initial power of the reactor was 20 MWth; it has been raised to 24 MWth in December 1967 and to 40 MWth in 1971 after several modifications (FORTISSIMO project).

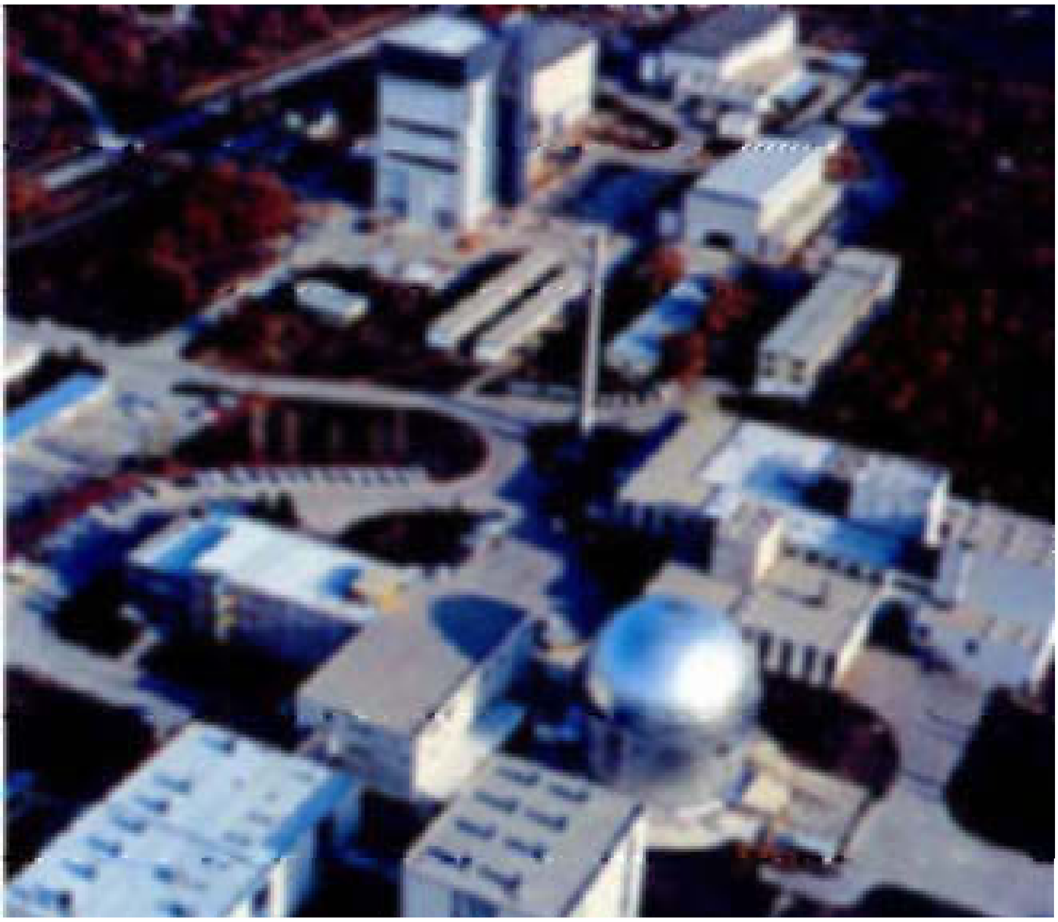


FIG. 1. General view of RAPSODIE.

FIG. 2. Cut view of the RAPSODIE reactor.

By the end of 1978, a small primary sodium leak was detected, which constrained the operation of the plant to about 22 MWth. In January 1982, another small sodium leak was detected in the nitrogen system (surrounding the primary vessel). The exact localization of the leak was estimated to be too long, too costly and too uncertain. Moreover, RAPSODIE had fulfilled its aims, it was then decided to finally shut down the reactor on 13th October 1982 after several end of life tests from May to October. The beginning of the first operations of the final shutdown started in April 1983. CEA was leader project in decommissioning of RAPSODIE reactor.

3.1.2. Different steps of the final shutdown of RAPSODIE [11]

The objective was to achieve on RAPSODIE a partial dismantling (stage 2 in IAEA classification). The main operations described chronologically to achieve this objective were the following:

- Removal of the assemblies from the basic nuclear facility;
- Removal of miscellaneous irradiated equipment and wastes contained in the interim storage wells (experimental devices, basic control devices,…);
- Isolating the reactor plant from contaminated systems;
- Washing and decontamination of the systems isolated from the reactor block.
- Dismantling the systems;
- Final containment of the reactor block;
- Setting of installations into safety configuration;
- Treatment of contaminated sodium;
- General sanitizing;
- These operations will be detailed in the following sections.

3.1.2.1. Defuelling

All the assemblies constituting the core of RAPSODIE have been extracted, cleaned and then stored in the pool. The process of cleaning the assemblies is made by a spray of water in a flow of inert gas (argon). This process was applied in a specific facility existing in every LMFR and called a cleaning pit [12]. This cleaning pit was used in normal operation when a fuel element was extracted before its inspection. All the fuel, breeder elements and control rods were removed and cleaned from April to November 1983. During the defuelling, sodium was static and was kept liquid by the residual power of the fuel elements. When this residual power became insufficient, pre-heating nitrogen circuit and safety cooling circuit were used to keep the sodium warm. The safety cooling system used to maintain the sodium in a liquid state was kept until the complete draining of the primary sodium in April 1984.

The other elements (metallic assemblies) were removed later. The 468 reflector assemblies constituting the core (222 made of nickel, 246 made of steel) were highly irradiated. They represented, in 1987, a global activity of about 130 000 Curies. In 1987 it was decided to take all the reflectors out and to store them at Cadarache site awaiting treatment before sending them to final repository. The operation of retrieving the reflectors from the vessel, washing them to eliminate traces of sodium, and installing them in a storage container lasted two years and required a workforce of 860 men per day with a production of 72 containers.

3.1.2.2. Draining of the sodium

The primary sodium was drained in the primary storage vessel in two steps. In the first step, the primary sodium coming from the primary pipes was drained, before the beginning of the defuelling. This operation took place in April 1983. In the second step the sodium remained in the primary vessel was transferred to the same sodium storage. The sodium was kept solid at room temperature under nitrogen cover gas.

After this draining and a further draining of two residual bulk of sodium, the amount of residual sodium let in the primary vessel was estimated to 70 kg of metallic sodium and 100 kg of aerosols and sodium oxides. An endoscopic examination of the primary vessel confirmed this first estimation. The primary vessel was kept under argon cover gas with a residual overpressure to prevent from any ingress of air.

In 1985, the sodium was transferred to another sodium storage. During this transfer, the primary sodium had a purification campaign from caesium (^{137}Cs and ^{134}Cs are the main radio contaminants in primary circuit of fast reactors). This purification campaign consisted in passing the liquid sodium through caesium traps. These caesium traps are made with carbonaceous solid material where the trapping of caesium is made by adsorption phenomenon [13, 14]. Thus the 37 tons of primary sodium of RAPSODIE was purified from around 1.85 1012 Bq of ^{137}Cs. The contamination of the primary sodium that was initially of 42 kBq/g of sodium was lowered to 5.8 kBq/g of sodium (reduction factor of more than 7).

The RAPSODIE secondary circuit was based on two circuits. Each of them contained 11 tons of sodium. This secondary sodium was drained in two storage vessels (15 m^3 each) and the sodium was kept solid at room temperature under a nitrogen cover gas. This operation took place in April 1983. This secondary sodium is still stored on the RAPSODIE site.

3.1.2.3. Cleaning of components

All the components used in RAPSODIE reactor (i.e. primary pumps, intermediate heat exchangers,…) were drained, removed, and cleaned in the cleaning pit devoted to the cleaning of components. In cleaning pits, residual sodium is progressively neutralized by the contact of a mist made of a mixture of droplets of demineralized water, nitrogen and carbon dioxide. Reaction of sodium with mist is creating sodium hydroxide and hydrogen. Aqueous sodium hydroxide produced by this reaction is then transformed into sodium carbonate in contact with carbon dioxide. Hydrogen is released in the ventilation duct after filtration and dilution. During all the chemical process, the cleaning pit is kept under inert gas (carbon dioxide or nitrogen).

The process is controlled by adjusting the density of water in the mist in function of the concentration of hydrogen released from the cleaning pit before dilution. In normal operation, this percentage of hydrogen is always lower than 1% [12, 15]. The components were then dismantled, cut and sent to the radioactive waste storage. The removal of control rod mechanisms and experimental devices present in the installation was a 12 months campaign, producing 32 containers. Primary cold traps and caesium traps were stored on the facility and will be sent and treated at the ATENA facility in 2007/2010. It is envisaged that the secondary sodium stored in two vessels will also be sent to this facility at the same time.

3.1.2.4. Treatment of the primary circuit [16]

After the draining of the primary sodium, the separation between the reactor vessel and its circuits was done. The objective was to tighten the main vessel. The following operations were carried out:

- Cutting the sodium inlet and outlet pipes and clad failure detection lines;
- Isolating the overflow tank from the reactor vessel;
- Isolating the sodium purification rack.

The primary circuit was then cleaned from sodium with an in situ treatment with the use of a heavy alcohol: ethylcarbitol. This washing was performed during the second quarter of 1988. The operation required the use of 4 200 litres of ethylcarbitol and 11 m^3 for various rinsing procedures. An activity of 4.2×10^{11} Bq in ^{137}Cs was eliminated from the surfaces then partially trapped on the ion exchanging resins.

FIG. 3. RAPSODIE sodium pipes.

The primary circuits were then decontaminated in three steps: alkaline washing to remove all the labile caesium, followed by acid (sulfo-nitric mixture) decontamination with CeIV to entrain the fixed contamination including about 10% of ^{60}Co, and a final phosphatation step. The estimated initial contamination level of 5500 Bq/cm^2 was reduced to less than 10 Bq/cm^2, allowing dismantling operations to proceed without constraints, producing waste material suitable for release and limiting an occupational dose estimated to 230 man.mSv. The decontamination operations generated 2600 kg of dry extract encapsulated by the local effluent treatment station. 164 m^2 were decontaminated after two effective working months with an average metal removal depth of 12 µm.

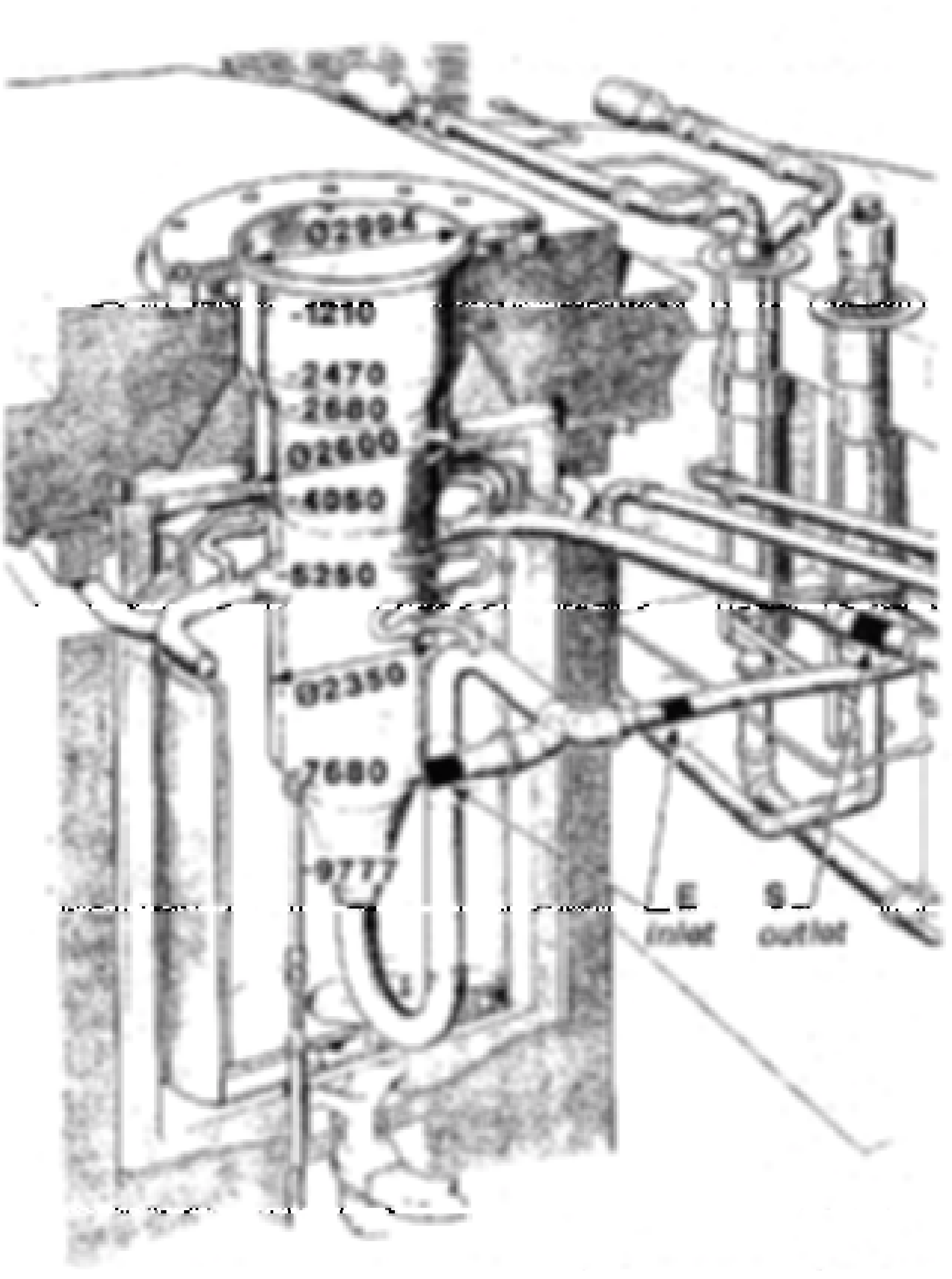


FIG. 4. Cut view of the primary circuit.

3.2.1.5. Reactor vessel containment

The main reason for confining the reactor block is to enclose all the activity in the plant within a limited solid structure. In addition, owing to the sodium residue in the unwashed vessel, that vessel must be constantly maintained in a nitrogen atmosphere in order to avoid reactions with atmospheric humidity. Thus, the reactor vessel has not been cleaned and is still with residual non-drainable sodium inside. This residual quantity of sodium is estimated to less than 100 kg. The cover gas is nitrogen. The containment of the reactor vessel is made of two barriers. The first barrier is made by plugging all the circuits of the vessel: inlet and outlet of sodium, gas circuit, purification circuit, pin failure detection circuit. The upper part of the reactor vessel including the rotating plugs was recovered by a welded metallic structure. The second barrier is the outer enclosure of the concrete made of rare earth (or Sercoter concrete) completed by a series of steel housing on the six sides of the reactor bloc. The cover gas of the first barrier is under nitrogen and over-pressurised.

3.2.1.6. Treatment of the secondary loops

RAPSODIE was constituted by two secondary loops. Heat removal was made by sodium/air heat exchangers. At the decision of the final shutdown of the reactor, the two loops were drained. One loop was kept and transformed in order to develop a technological program to test a sodium pump (CARUSO program), it was then dismantled in the 90's. The other loop was quickly dismantled. The two circuits were dismantled without preliminary in situ treatment. Pipes were cut and then treated with water in specific facility held at Cadarache. Global activity in tritium of the secondary sodium was estimated at the date of 1st January 1994, to 2.21 GBq/t. The activity of the residual sodium in the loop was estimated to 15 Bq/g of sodium.

3.1.2.7. Dismantling of the primary system and its auxiliary systems

The pipes and tanks were cut using the plasma torch except in "high risk" areas where the cutting was performed using a saw or chain saw. All the components of the primary system (pipes, tanks) were previously drawn, in order to define the cutting line, and were individually marked to facilitate future radiological identification of every waste produced. The dismantling operation produced 512 components, the largest dimensions did not exceeded one metre. The weight of waste produced by the dismantling operation was 13.472 kg of stainless steel. The cutting of the primary system required 650 metres of plates and pipes, between 3 and 12 mm thick, involving a production rate of 12.75 m per day per man.

3.2.1.8. Primary sodium treatment

For the treatment of primary sodium, CEA has developed a process called NOAH process in order to continuously transform sodium in sodium hydroxide. The principle of NOAH process consists in injecting small quantities of liquid sodium by a dosing pump through a sodium nozzle into a strong flow of aqueous sodium hydroxide (concentration of 10 mol/L) flowing within a closed vessel. The liquid sodium is scattered in the water and reacts smoothly and continuously. The chemical reaction being exothermic, it requires a continuous cooling through a liquid/liquid heat exchanger. The hydrogen produced by the chemical reaction is filtered, dried and diluted before release to the stack. The aqueous sodium hydroxide concentration is monitored and adjusted by adding water. The pilot facility to validate the NOAH process was developed by CEA between 1985 and 1989 [17, 18]. Then the design of the DESORA facility (DEstruction du SOdium de RApsodie) started in 1989 and was carried out by Framatome (Fig. 5).

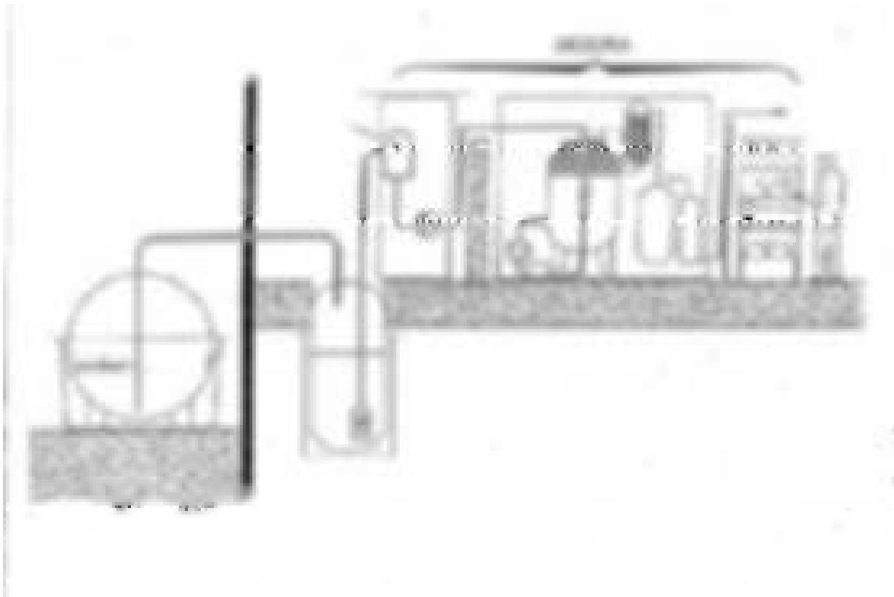


FIG. 5. The DESORA facility.

Sodium treatment by the DESORA facility started in 1994 and allowed the treatment of the 37 tons of primary sodium in three months (nominal flow rate of 40 kg/h) [19, 20]. The treatment of the 37 tons of sodium produced around 180 m^3 of concentrated sodium hydroxide that was used to neutralize radioactive acidic effluents at La Hague reprocessing facility.

3.1.2.9. In-situ cleaning of a tank and its consequences [21]

On 31 March 1994, during the cleaning of the residual sodium contained in a tank located in a hall outside the containment building of the RAPSODIE reactor, an explosion occurred. One member of the CEA staff was killed and four people were injured. The sodium present in the tank in which the accident occurred comes from the primary cooling circuits of RAPSODIE. This residual sodium was constituted by the undrainable sodium remained at the bottom of the tank at the end of the DESORA campaign. Before being dismantled, the tank had to be cleaned in order to remove the residual sodium. The process selected to perform this clean up operation, already implemented several times [22], consisted in progressively introducing in the tank a heavy alcohol called ethylcarbitol, while monitoring the reaction through temperature, pressure, hydrogen and oxygen measurements. The major cause of the accident was due to the formation of an heterogeneous physical-chemical environment, complex and multiphasic made of three basic components: alcohol, alcoholate and sodium. This environment turned out to be particularly favourable to the development of thermal decomposition reaction and/or catalytic exothermal reactions. Large quantities of gases (including hydrogen and light hydrocarbon compounds) were thus produced. Shortly after the last alcohol injection on 31 March, the phenomenon run out of control, leading to a sudden rupture of the overpressurised tank, then to the explosion of the gases mixture blown out in the hall. After this accident, a commission of enquiries was set up. The complicate chemical structure of ethylcarbitol ($C_6H_{14}O_3$) was recognized to facilitate the possibility of thermal runaways. The immediate recommendation of this commission was to forbid the use of ethylcarbitol or other heavy alcohol in the treatment of sodium. The same kind of accident occurred two years later in Germany [23].

3.1.2.10. Activity and dose rate assimilated

The removed activity is estimated to around 4 800 TBq. The activity contained in the primary vessel was estimated to 600 TBq in 1990 (mainly ^{60}Co). The dose assimilated by the whole personnel having worked from 1987 to 1994 on the installation is 224 mSv. The dose assimilated during the year 1988 was 117 mSv, due mostly to the work of separating the reactor vessel from the primary system before the latter was washed and decontaminated.

3.1.2.11. Present situation of the RAPSODIE reactor and future

After the accident, the main activity on the reactor was to rebuild the buildings and repair the damages. The objective is to reach the stage 2 of the IAEA decommissioning phase. Then the surveillance state should last from 2005 to 2020 before the final decommissioning of the reactor (stage 3) [24]. A project is in progress to estimate the cost benefits that can be gained by reducing the schedule of the decommissioning of this reactor.

3.2. SUPERPHÉNIX reactor

3.2.1. History and general description of the reactor [2, 25 - 28]

3.2.1.1. History

As early as 1971, the French Atomic Energy Commission (CEA) working closely with EDF carried out preliminary studies for the new reactor. In 1973, an agreement was signed by EDF (France), ENEL (Italy) and RWE (Germany) setting up NERSA: a limited company to operate a fast breeder reactor. NERSA acted as owner and operator with the latter role being entrusted to EDF. The creation of NERSA was confirmed by a decree passed in 1974. Two companies, Novatome and NIRA (today FRAMATOME ANP and ANSOLDO) performed jointly the design and the construction of the reactor as prime contractor. In 1977 started the plant construction at Creys Malville near the Rhône river (Fig. 6).

FIG. 6. General view of SPX buildings.

In October 1984, the first filling with sodium was successfully completed. SUPERPHÉNIX went first critical in September 1985 and then followed, after the first connection to the grid on 14 January 1986, and reached the nominal power (1200 MWe) in December 1986. During the operation of SUPERPHÉNIX, two major events occurred: a sodium leak was discovered in March 1987 in the wall of the main vessel of the in sodium assemblies storage vessel. The treatment of the event and the appropriate remediation lasted 20 months.

The second event occurred in July 1990: A microleak in the diaphragms of the compressor in the argon circuit leaded to an air ingress in the primary circuit and consequently to a sodium pollution beyond the specified limits. The purification finished in spring 1991. Then started administrative procedure that allows to re-start the reactor in 1994 (three years later). The plant operated throughout the whole year in a very satisfactory manner up to 90% of its rated power. On 24th December 1996, the plant was voluntarily shut down for the ten years inspection on steam generators (regulatory inspections). At this time the operation time was equivalent to 320 Equivalent Full Power Days. It will never restart. Indeed, in June 1997, the French Prime Minister announced the intention to abandon the plant, and on the 2nd February 1998, the decision of its definitive shutdown was announced by the Government.

To sum up the life of SUPERPHÉNIX operation:

- 53 months (40% of the total time), the plant operated normally generating 7.9 billion of kWh;
- 25 months of shutdowns needed for modifications of the plant or reworking as a consequence of major events (sodium leak and sodium pollution);
- 54 months of shutdowns imposed by administrative procedures, public inquiries, etc…

3.2.1.2. General description of the SUPERPHÉNIX reactor

SUPERPHÉNIX is a pool type liquid metal fast reactor. The coolant is sodium. It is composed of a main reactor vessel of 21 m diameter and 19.5 m high filled with 3250 tons of liquid sodium (Fig. 7).

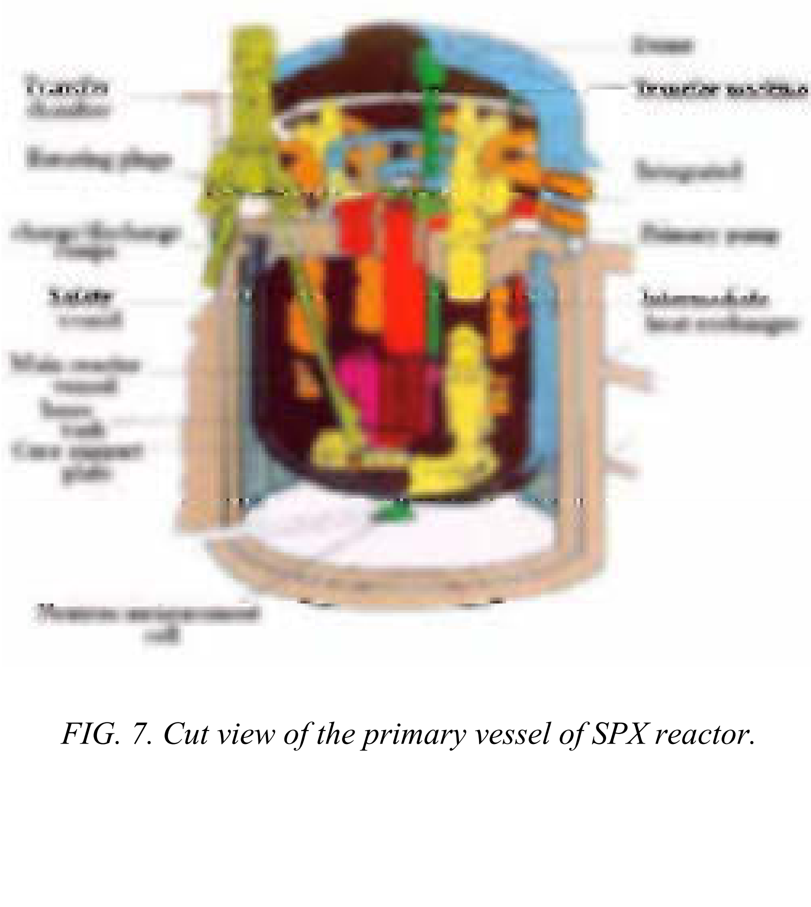


FIG. 7. Cut view of the primary vessel of SPX reactor.

The primary system includes four primary pumps and eight intermediate heat exchangers connected with four secondary loops. The total amount of sodium in these four secondary loops is 1500 tons. The core of SPX is made of fuel elements (mixed UO_2-PuO_2), breeder elements, rods plus complementary control rods (pins containing pellets of boron carbide B_4C). The power of SPX reactor was 3000 thermal MW converted to 1242 electric MW. At the date of the decision of the final shutdown of the reactor, SPX was in operation only 320 Equivalent Full Power Days. During this time there was no cladding failure. Therefore it can be assumed that SUPERPHÉNIX has a very low residual activity and minor contamination in the primary sodium. Thus the residual activity of the primary sodium was measured. The values in 2000 were:

- ^{22}Na: 3500 Bq/g (period 2.6 y);
- ^{137}Cs: < 5Bq/g (period 30 y);
- ^{3}H: between 5000 and 20000 Bq/g (period 12.6 y);
- ^{54}Mn: 5 Bq/g (period 312 d).

3.2.2. Scenario for the definitive shutdown of SUPERPHÉNIX

The first definitive shutdown decree was published on 30th December 1998 [29]. This decree authorizes the following operations:

- Core unloading;
- Draining of sodium and its storage;
- Dismantling of non-nuclear installations definitively out of order and/or unnecessary for safety.

The subsequent phases, including sodium treatment will be subject to a further decree and authorizations. Taking into account this decree, EDF with its industrial partner Framatome ANP, studied a global scenario for the general decommissioning of the plant. This scenario takes into account the experimental feedback of the decommissioning of several LMFRs (RAPSODIE, PFR) and is of course adapted to the specificities and particularities of the Superphénix plant. The general strategy of EDF is to achieve a voluntarist scenario for the global decommissioning of the reactor. The different steps of this decommissioning and the global strategy will be now seen in the Sections 3.2.2.1 – 3.2.2.9.

3.2.2.1. Unloading

The core unloading has started at the end of 1999 and it is planned to be achieved by the end of 2002. It consists in the removal of:

- 368 fuel assemblies;
- 229 breeder assemblies;
- 50 in-core absorber subassemblies;
- 3 neutron guides.

Then steel assemblies will be removed, that is to say:

- 184 steel reflector subassemblies;
- 1076 lateral neutron protections.

The unloading of every element starts with a cleaning process to remove the residual sodium located of every component (estimated to a maximum of 600 grams of sodium per element). The cleaning process employed is globally the same concept as the RAPSODIE cleaning process. These elements are then to be stored under water in the spent fuel pool of SPX plant. The unloading of the core is done without the use of dummy core. In 2001, the defuelling was in operation. The lateral neutron protections are too numerous to be stored in the pool. A lot of them will be sent to the medium activity storage center (CSA Centre de Stockage de L'Aube), those which are too active have to be sent will be temporarily stored on site in containers.

3.2.2.2. Maintaining sodium in temperature

As the decay heat is lower than reactor thermal loss, the sodium temperature must be maintained by an outside heat source. In normal operation, the primary sodium can be kept at a given temperature by the secondary loops or by the primary pumps. During core unloading, the primary pumps must be shut down. During draining of the reactor vessel, when sodium level drops below the input window of the intermediate heat exchangers, heating through the secondary loops will no longer be possible. Thus, it was decided to install a system to maintain the temperature by electric tracing cables, fixed on the safety vessel, and then to thermally insulate this vessel. It was the first operation done on the SPX vessel. The primary pumps were definitively stopped in September 1999,

FIG. 8. Operators putting tracing cable on the SPX safety vessel.

3.2.2.3. Sodium draining

The secondary loop has been drained off and the sodium is stored in solid state in the storage tanks. In a first scenario, it was also decided to drain the sodium from the primary vessel to an interim storage before its conversion into sodium hydroxide based on the NOAH process. Finally it has been decided that the primary sodium will not be drained, but transferred on line to the SPX sodium disposal plant. This treatment operation should last four years and should start around 2006. Thus the draining of the primary vessel will be done very progressively.

3.2.2.4. Sodium retention treatment [30, 31]

When the primary vessel is drained, volumes of liquid metal will remain trapped by internal structures. These trapped volumes are estimated to 1% of the initial amount of sodium in the primary vessel that is to say about 37 tons. The residual sodium is mainly trapped in three areas:

- 11.6 m^3 are trapped in the core catcher which is made of two large piled plates;
- 12.2 m^3 are trapped in the lower part of the core diagrid;
- 8.6 m^3 are trapped at the bottom of the main vessel reactor.

To remove these main retentions, several techniques were selected. Thus sodium trapped in the lower part of the core diagrid will be siphoned and sodium trapped at the bottom of the main reactor will be pumped by an adapted device. The core catcher will be drilled in order drain the trapped sodium from this hole to the bottom of the vessel and then to pump it out. After this complementary draining, the sodium remaining in the reactor vessel is estimated to 2.6 tons (1.3 tons as films of sodium and 1.3 tons as bulks of sodium). Studies are carried out to evaluate the possibility to reduce these values by local draining or sucking for non-prohibitive costs.

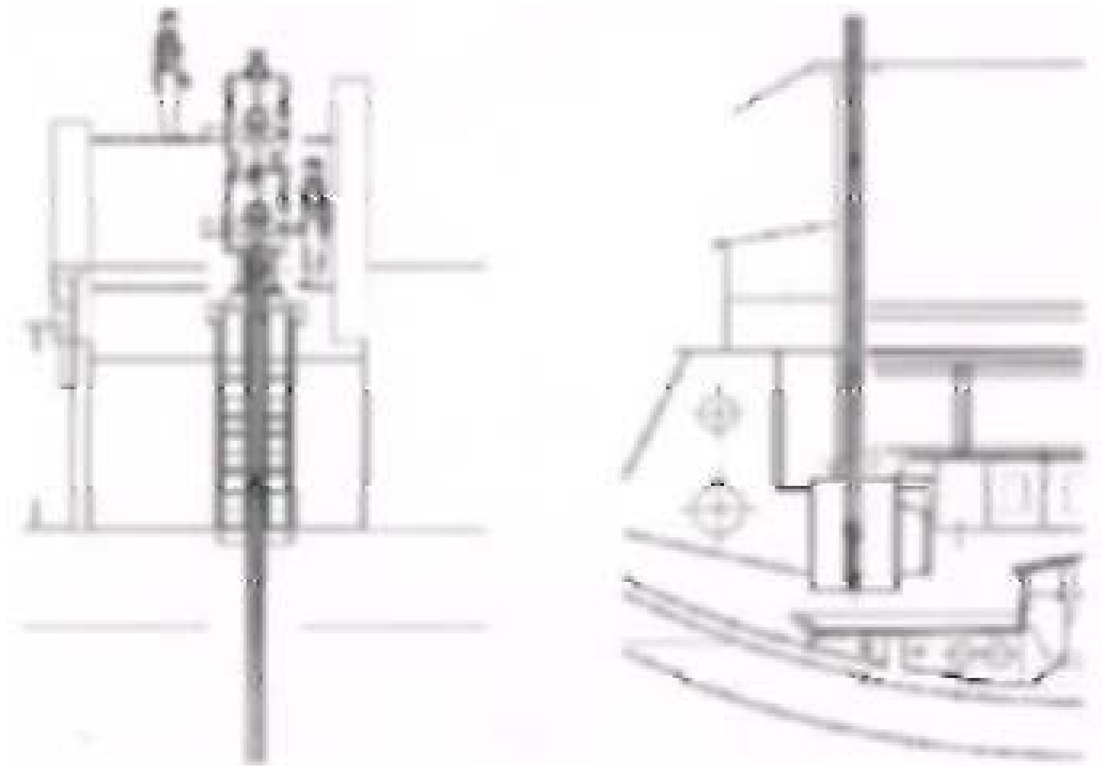


FIG. 9. Methods to allow a further draining of the SPX primary vessel.

3.2.2.5. Sodium treatment [40]

The reuse of this sodium for nuclear operation was not possible due to the absence of building of new LMFR in the near future. Thus the transformation of metallic sodium to a non reactive material was decided. To do so, it was decided to use the reference NOAH process that will transform metallic sodium to concentrated sodium hydroxide (10 mol/L). The treatment of primary sodium (3250 tons) and secondary sodium (1500 tons) will produce large amount of sodium hydroxide (19 000 tons) or neutralized salts such as sodium sulphate (25 000 tons). The facility envisaged will be based on the design of the SDP facility (Sodium Disposal Plant) built to treat the primary and secondary sodium of PFR [30, 32]. The estimated treatment flow rate will be 6 metric tons per day. The final destination of this by-product must be clearly defined and authorized by a decree. Two major solutions can be envisaged:

- Release of sodium sulphate into the Rhône river. Considering the low activity of the primary sodium and absence of fission products (no fuel clad failures), release into the Rhône river of the equivalent of 2.5 tons of sodium treated per day is possible within the framework of current release authorizations (250 GBq/year of liquid effluent excluding tritium). Studies of the environmental impact have been carried out, they have shown it would be very low (0.1 mSv); and
- Fabrication of sodium hydroxide based cement before conditioning under the form of concrete blocks to be stored as Very Low Radioactive Waste at CSTFA disposal facility (Centre de Stockage TFA de l'Aube) under the control of ANDRA (Agence Nationale de gestion des Déchets Radioactifs), the French national radioactive waste control agency.

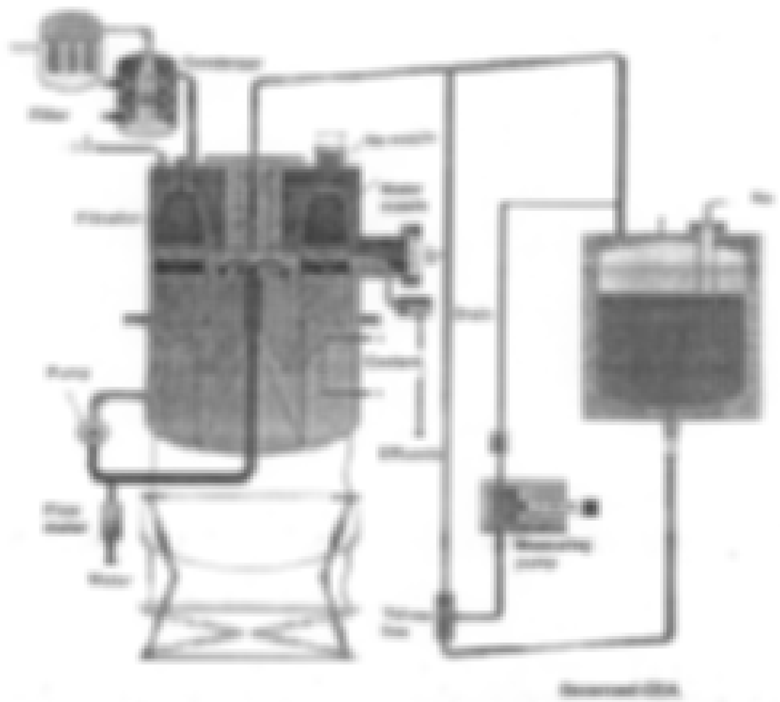


FIG. 10. The NOAH process.

For that second technical option, several tests were done to evaluate the possibility to stabilize sodium hydroxide or sodium salts into concrete. These tests were successful [33]. The EDF have chosen the cementation of the sodium hydroxide because the other solution was too hazardous in terms of time schedule.

3.2.2.6. Treatment of structures [39]

Studies are carried out to specify treatment of:

- Small and large primary components such as primary pumps, intermediate heat exchangers;
- Structures after sodium draining;
- Secondary loops.

For small and large components the use of the SPX cleaning pits will be necessary to eliminate the residual sodium remained after draining. The process used in the cleaning pits is a mist spraying. This mist is composed by demineralized water, carbon dioxide and nitrogen. For the primary vessel and secondary loops, it is intended to apply a carbonation method.

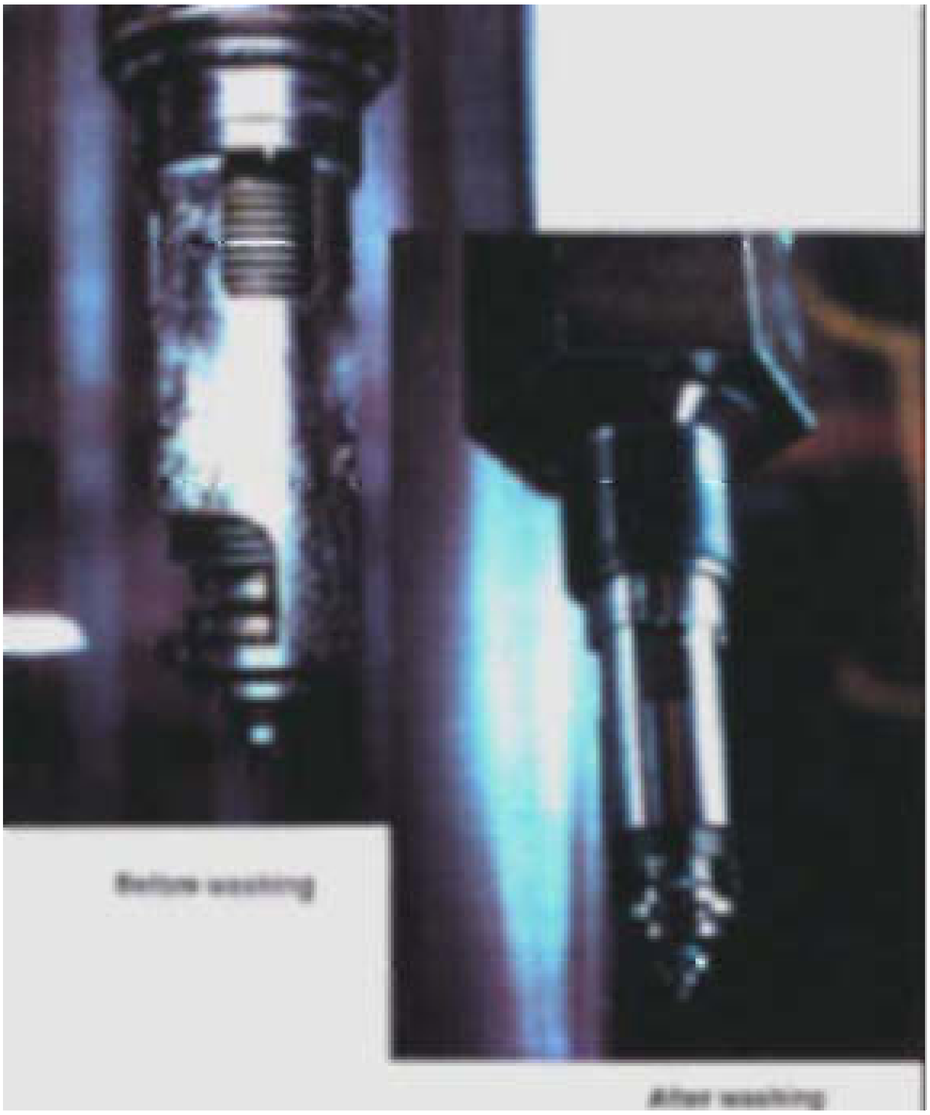


FIG. 11. A component before and after cleaning.

FIG. 12.

FIG. 13.

FIGS 12. and 13. Pictures of the internal of the SPX spent fuel storage vessel after the carbonation treatment.

It consists in injecting inside the component or the vessel a circulation gas carrying a very low amount of water: the gas must be kept under the saturation point in water. The circulation gas is a mixture of carbon dioxide and nitrogen. Therefore in contact with the humidity carried by the gas, sodium is smoothly reacting, producing anhydrous sodium hydroxide that will be then transformed to solid sodium bicarbonate ($NaHCO_3$) by the contact with CO_2. This process has already been applied with success for the internal cleaning of residual films of sodium of the spent fuel storage vessel of Superphénix in 1988 before its dismantling [34]. The objective is now to define kinetics and a global process to allow the treatment of greater bulks of sodium [35]. The application is the treatment of the residual sodium retained in the primary vessel (after the draining of the sodium and the treatment retention operation) as far as it is reasonably possible.

3.2.2.7. Radiological inventory

Ruling imposes drawing up an inventory of current on-site radioactivity in view of its decommissioning. Studies are committed for:

- Drawing up the inventory and localization of fission products, contamination, and activation;
- Calculating primary circuit structural dose rates after sodium draining;
- Calculating components dose rates;
- Estimating the activity at every decommissioning step.

This radiological inventory is made by on site measurements, calculation and estimation of residual activity by the mean of computing code, sampling and radiochemical analyses in order to verify the computational results.

3.2.2.8. Sodium waste treatment

A major sodium waste treatment issue will come from the decommissioning of the secondary loops. For this operation, two alternatives are currently studied: in-situ water flushing after carbonation treatment or washing of metallic pieces after dismantling in a specific facility based on water spraying process. Majority of components extracted from the primary circuit will be treated by the mean of the cleaning pits. Some components which contains sodium or NaK alloy and which require specific treatment will probably be treated in a CEA facility called ATENA that will be constructed in 2007. This is particularly the case for primary and secondary cold traps. These components present the major drawbacks to be not self-draining and to concentrate sodium impurities (sodium oxides and sodium hydride) and contamination (mainly caesium for primary cold traps and exclusively tritium for secondary cold traps). Methodology for processing these kinds of components has been developed by CEA [8, 36], and will be applied at Marcoule, on ATENA based nuclear installation [37].

For components containing NaK alloys, specific treatment will be defined with respect to the particular specificity of NaK handling and treatment especially when the NaK can be supposed oxidized [38].

3.2.2.9. Decommissioning level 3 [41]

EDF's strategy is now to shorten the time between the level 2 and the level 3. Thus it is envisaged to achieve the level 3 decommissioning in 2025. To do so, a major part of the work will be focused on the dismantling of the primary vessel and dismantling of all the primary components. The studies on this subject are already engaged. First options oriented between different technical options. One of them is, after a possible cleaning of residual sodium and carbonates to fill the primary vessel with water and to realize under water cutting of the internal structures.

4. GENERAL CONCLUSIONS ON LMFRS DECOMMISSIONING

4.1. Recommendations on LMFRs decommissioning

Decommissioning of LMFRs is still a young technology that is not completely mature: it is not possible to say at this present time that a LMFR has reached a stage 3 decommissioning level. It is then necessary to gain the maximal experimental feedback from the passed realization and from the decommissioning operations currently in progress. Thus from several works done in France for RAPSODIE and Superphénix and also from international technical exchange with other countries that are involved in LMFR de commissioning the following recommendations can be highlighted:

- Prepare the work and the strategy of decommissioning in advance:
  When it is possible, it is worth to start the decommissioning studies at least two years before the date of the final shutdown. This type of work will be necessary to define the general decommissioning strategies and to define in advance what will be the main works to start, the possible supporting R&D to develop and to identify where will be the major difficulties that can provoke major risks in the schedule.
- Use proven technologies:
  At a decommissioning stage, as far as it is possible, it is better to use proven technologies for all the operation linked to the decommissioning. Indeed, the time of decommissioning is not a time of important development and the main objective is to reduce the costs. These objectives are not compatible with the development of known technologies or the start of new field of competence. It is better to know what are the general techniques used all over the world in the frame of LMFRs decommissioning, to compare them and to adapt them to the specificity of the reactor to dismantle.
- Use simple chemistry to treat sodium (water):
  For the treatment of sodium, the easier process is the better. Among the wide range of processes developed to treat sodium, water has proven to be the most efficient reactant because: water is cheap, it produces well known by products (sodium hydroxide and hydrogen), it has been widely used to treat sodium. In some particular cases the ignition of fire can be a way to treat metallic sodium. In that case, the treatment of contaminated gaseous effluents must be particularly well defined. All the other products that can be used to treat sodium (organic compounds, oil, ammonia) will induce much more drawbacks than advantages.
- Do the decommissioning as fast as we can at the beginning:
  This is of major importance in order to keep the exploitation team in place as far as possible and to be very efficient the first years of the decommissioning program where the most difficult operations will take place: defuelling, sodium draining and sodium treatment. Knowledge of the plant disappears very quickly due to several reasons: retirement of the staff, transfer of operators to other plants… So it is necessary to do all the possible works immediately.

Moreover, this strategy has the advantage to keep on the team a new challenge where a new motivation can be found.

To prevent from the loss of knowledge, in parallel development of a knowledge preservation program must be done. This knowledge preservation will cover the following items: data recording, written synthesis of specific subject, recording of the general knowledge of experts before their retirement, preservation of the training to sodium and NaK technologies for the new teams. This strategy of knowledge preservation is in progress in France [42, 43].

4.2. Conclusions

As it can be seen from this paper, the decommissioning of a LMFR necessitates a specific knowledge on sodium and NaK handling until the least gram of alkali metal has disappeared of the plant. Thus the decommissioning of a fast reactor becomes a challenge where the knowledge preservation of sodium technology during the whole decommissioning time is the most difficult part to achieve.

Nevertheless, even if the sodium technology is very specific, the decommissioning of Fast Reactors with sodium coolant does not present major technical difficulties. It can be seen from the dismantling of RAPSODIE and Superphénix that every step of decommissioning has already its identified technical solution that had already been tested. Decommissioning a LMFR is a long and huge work but all the steps are technically feasibles. In this field, the premature decommissioning of Superphénix will place French technology in the first places.

REFERENCES

[1] BARRE, B., Fast breeders: Back to the future!, Proc. World Nuclear Congress and Expo (ENC'98) Trans., Nice, France, 25-28 October 1998, Vol. IV, Workshops, (1998) pp. 70-74.

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of liquid metal cooled fast breeder reactors, Technical reports series n°246, Vienna (1985).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Fast reactor Database, IAEA-TECDOC-866, Vienna (1996).

[4] VENDRYES, G., Les Surgénérateurs, Collection du Commissariat à l'Energie Atomique, Novembre 1986.

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of liquid metal cooled fast reactors technology, TECDOC-1083, Vienna (1999).

[6] Atomic Energy Commission, Liquid - Metals Handbook. SODIUM - NaK supplement, TID 5277, Department of the Navy, Washington D.C., (1955).

[7] BORGSTEDT, H.U., MATHEWS, C.K., Applied chemistry of the alkali metals, Plenum press, New York (1987).

[8] RODRIGUEZ, G., SAROUL, J., ARNAULD DES LIONS, J.P., Methods for sodium waste treatment coming from liquid metal fast reactors, Proc. World Nuclear Congress and Expo (ENC'98) Trans., Nice, France, 25-28 October 1998, Vol. IV, Workshops, (1998) pp. 85-89.

[9] LECLERE, J., Fast reactor development program in France in 1999, IAEA-TC-385.72 International Atomic Energy Agency, Vienna (2000).

[10] DENIELOU, G., et al., RAPSODIE development program: from design to operation, Proc. Natl. Top. Mtg., San Francisco, 1967, American Nuclear Society, Rep. ANS-101, San Francisco, USA (1967) pp. 55–75.

[11] ROGER, J., Stage 2: Dismantling of reactor. Case of the experimental FBR RAPSODIE, paper presented in the International Symposium on Decontamination and Decommissioning, Knoxville, USA, 1994.

[12] LUTTON, J.M., COLBURN, R.P., WELCH, F., Sodium removal and decontamination of LMFBR components for maintenance, Atomic Energy Review n°184 (1980).

[13] HANEBECK, N., MSIKA, D., MISRAKI, J., ALLEGRE, R., TUSCHE, R., Development of Cs traps for commercial sodium-cooled fast breeder reactors, ELCESNA, KNK II, RAPSODIE, Proc. Intl. Conf. Liquid Metal Engineering and Technology (LIMET 88), Avignon, France, 17-21 October 1988, organized by Societe Francaise d'Energie Nucleaire.

[14] KRASNOYAROV, N.V., POLYAKOV, V.L., SOBOLEV, A.M., The removal of cesium from the fast primary coolant, Proc. 3rd Intern. Conf. Liquid Metal Engineering and Technology (LIMET 84) Oxford, The British Nuclear Society, London, (1984) pp. 185-190.

[15] MASSE, F., RODRIGUEZ, G., Cleaning and decontamination: experimental feedback from PHÉNIX, paper presented in the IAEA Specialists' Meeting on Sodium Removal and Disposal from LMFRs in Normal Operations and in the Framework of Decommissioning, Aix-en-Provence, France, 1997.

[16] COSTES, J.R., ANTOINE, P., GAUCHON, J.P., Decontamination before dismantling a fast breeder reactor cooling system, paper presented in the IAEA Specialists' Meeting on Sodium Removal and Disposal from LMFRs in Normal Operations and in the Framework of Decommissioning, Aix-en-Provence, France, 1997.

[17] LEVILLAIN, C., LAFON, A., NOAH Experimental plant for treating contaminated sodium, International patent no. 8606266, 1986.

[18] LEVILLAIN, C., MICHAILLE, P., Experimental plant for treating contaminated sodium, Proc. 4th Intl Conf Liquid Metal Engineering and Technology (LIMET 88), organized by Societe Francaise d' Energie Nucleaire 17-21 October 1988, Avignon, France (1988).

[19] de SEROUX, N., LATGE, C., ROGER, J., Contaminated sodium disposal, NOAH process, Proc. Intl. Conf. On Safe Management and Disposal of Nuclear Waste (SAFEWASTE 1993), Avignon, France, 13-18 June 1993, Sociéte Française d' énergie nucléaire (SFEN), Paris (1993).

[20] ROGER, J., LATGE, C., RODRIGUEZ, G., Transformation of sodium from the RAPSODIE Fast Breeder Reactor into sodium hydroxide, paper presented in the International Conference on Decommissioning of Nuclear Installations, Luxembourg, 1994.

[21] MARMONIER, P., DEL NEGRO, R., Information about the accident occurred near RAPSODIE (1994, March 31st), IAEA Specialists' Meeting on Sodium Removal and Disposal from LMFRs in Normal Operations and in the Framework of Decommissioning, Aix-en-Provence, France, 1997.

[22] ALLEGRE, R., OLLIVIER, J., MALET, J.C., MAGAUD, G, HANEBECK, N., TUSCHE, R., EBSEN, W., Expériences de nettoyage à l'éthylcarbitol de réservoirs et de circuits en sodium, Proc. 4th Intl Conf Liquid Metal Engineering and Technology (LIMET 88), organized by Societe Francaise d' Energie Nucleaire 17–21 October 1988, Avignon, France (1988).

[23] MINGES, J., CHERDRON, W., SCHUTZ, W., Short report of an accident during sodium cleanup with ethyl carbitol in a storage tank of a research facility, IAEA Specialists' Meeting on Sodium Removal and Disposal from LMFRs in Normal Operations and in the Framework of Decommissioning, Aix-en-Provence, France, 1997.

[24] CERDAN, G., Le démantèlement des installations nucléaires expérimentales du CEA, Revue Générale Nucléaire - RGN n°6, November-December 2000 (2000) pp. 66–68.

[25] VENDRYES, G., SUPERPHENIX, pourquoi?, Nucléon Editor, 1997.

[26] TAICLET, G., Superphénix: the story of a premature end, Revue Générale Nucléaire, RGN 1998 n°4, Août-Septembre 1998 (1998) pp. 114–122.

[27] CARLE, R., What lessons are to be learned from the Superphénix saga?, Nuclear Europe Worldscan, n°5-6 1998 (1998) pp. 47–49.

[28] LECLERE, J., GIRAUD, B., HUBERT, G., LEFEVRE, J.C., Thirty years of fast reactor development in France, Proc. World Nuclear Congress and Expo (ENC'98) Trans., Nice, France, 25-28 October 1998, Vol. IV, Workshops, (1998) pp. 31–39.

[29] DECREE n°98-1305, 30th December 1998, French Official Journal n°303 published the 31st December 1998.

[30] RODRIGUEZ, G., MASSE, F., ROBIN, J.C., BERTE, M., BUCHWALDER, M., SERPANTIE, J.P., Sodium waste treatment: from process development to industrial application, Proc. Intl. Conf. On Safe Management and Disposal of Nuclear Waste (SAFEWASTE 2000), Montpellier, France, 1–5 October 2000, Sociéte Française d' énergie nucléaire (SFEN), Paris, Vol 1, (2000) pp. 346-356.

[31] BAQUE, F., HAUDIQUET, C., MAJOR, P., Superphénix LMFR decommissioning. Sodium retention treatment by mechanical drilling of internal structures, Proc. 8th International Conference on Radioactive Waste Management and Environmental Remediation (ICEM'01), Bruges (Brugge), Belgium, 30 September – 4 October 2001, sponsored and organized by American Society of Mechanical Engineers (ASME) et. al.; in co-operation with the U.S. Department of energy (USDOE) et al., ANIBAL TABOAS, RIK VANBRABANT, GARY BENDA (Eds), New York, New York: ASME, c2002. TD898.14.M35 I57 2001

[32] de MAGNY, E., BERTE, M., Fast reactor bulk sodium coolant disposal. NOAH process application, paper presented in the IAEA Specialists' Meeting on Sodium Removal and Disposal from LMFRs in Normal Operations and in the Framework of Decommissioning, Aix-en-Provence, France, 1997.

[33] RODRIGUEZ, G., CAMARO, S., FIQUET, O., BERNARD, A., LE BESCOP, P., Study of optimal transformation of liquid effluents resulting from the destruction of radioactive sodium by water into ultimate solid wastes, paper presented in the IAEA Specialists' Meeting on Sodium Removal and Disposal from LMFRs in Normal Operations and in the Framework of Decommissioning, Aix-en-Provence, France, 1997.

[34] KERVEGANT, M., POMIE, P., VERDIER, P., FOURNIE, J.L., DAVIOT, J., Nettoyage de la cuve et de l'intercuve du barillet de SUPERPHENIX Proc. Intl. Conf. Liquid Metal Engineering and Technology (LIMET 88), Avignon, France, 17-21 October 1988, organized by Societe Francaise d'Energie Nucleaire.

[35] RODRIGUEZ, G., GASTALDI, O., Sodium carbonatation process development in a view of treatment of the primary circuit of Liquid Metal Fast Reactor (LMFR) in decommissioning phases, Proc. 8th International Conference on Radioactive Waste Management and Environmental Remediation (ICEM'01), Bruges (Brugge), Belgium, 30 September – 4 October 2001, sponsored and organized by American Society of Mechanical Engineers (ASME) et. al.; in co-operation with the U.S. Department of energy (USDOE) et al., ANIBAL TABOAS, RIK VANBRABANT, GARY BENDA (Eds), New York, New York: ASME, c2002. TD898.14.M35 I57 2001.

[36] RODRIGUEZ, G., GOEDERT, N., COURNILLE, M., Development of cutting techniques applied on sodium wastes. Application to Liquid Metal Fast Reactors decommissioning, Proc. 8th International Conference on Radioactive Waste Management and Environmental Remediation (ICEM'01), Bruges (Brugge), Belgium, 30 September – 4 October 2001, sponsored and organized by American Society of Mechanical Engineers (ASME) et. al.; in co-operation with the U.S. Department of energy (USDOE) et al., ANIBAL TABOAS, RIK VANBRABANT, GARY BENDA (Eds), New York, New York: ASME, c2002. TD898.14.M35 I57 2001.

[37] DEMOISY, Y., THOMINE, G., RODRIGUEZ, G., ATENA, Project of storage and disposal plant for radioactive sodium wastes, Proc. 8th International Conference on Radioactive Waste Management and Environmental Remediation (ICEM'01), Bruges (Brugge), Belgium, 30 September – 4 October 2001, sponsored and organized by American Society of Mechanical Engineers (ASME) et. al.; in co-operation with the U.S. Department of energy (USDOE) et al., ANIBAL TABOAS, RIK VANBRABANT, GARY BENDA (Eds), New York, New York: ASME, c2002. TD898.14.M35 I57 2001.

[38] DESREUMAUX, J., CALAIS, M., ADRIANO, R., TRAMBAUD, S., KAPPENSTEIN, C., NGUEFACK, M., Reactions of sodium-potassium alloys with inert gas impurities. Potential hazards after oxidation, Eur. Jour. Inorg. Chem. (2000) pp. 2031-2045.

[39] CHIAROT, D., JOULIA, E., Strategy for the carbonatation of the SUPERPHÉNIX sodium circuits and reactor block, paper presented in the IAEA Technical Meeting on Operational and Decommissioning Experience with Fast Reactors, Cadarache, France, 11-15 March 2002, to be published as IAEA-TECDOC.

[40] ALPHONSE, Ph., Comprehensive strategy for the dismantling of SUPERPHÉNIX plant, paper presented in the IAEA Technical Meeting on Operational and Decommissioning Experience with Fast Reactors, Cadarache, France, 11–15 March 2002, to be published as IAEA-TECDOC.

[41] CHIAROT, D., SPX: Strategy and orientations for dismantling the reactor bloc. State of reflections in February 2002, paper presented in the IAEA Technical Meeting on Operational and Decommissioning Experience with Fast Reactors, Cadarache, France, 11-15 March 2002, to be published as IAEA-TECDOC.

[42] BAQUE, F., R&D LMFRs knowledge preservation French project, paper presented in the IAEA Technical Meeting on Operational and Decommissioning Experience with Fast Reactors, Cadarache, France, 11–15 March 2002, to be published as IAEA-TECDOC.

[43] RODRIGUEZ, G., R&D LMFRs knowledge preservation French project: application to sodium coolant and cover gas, paper presented in the IAEA Technical Meeting on Operational and Decommissioning Experience with Fast Reactors, Cadarache, France, 11-15 March 2002, to be published as IAEA-TECDOC

OVERALL STRATEGY OF CREYS MALVILLE POWER STATION DISMANTLING

P. ALPHONSE
EDF/CIDEN, Creys Malville, France

Abstract

The power station was stopped by a government decision following the elections in 1997. This shutdown was then officialized by a letter dated April 1998 and the decree of December 1998. This was a non-technical shutdown and as such had not been envisaged; there has been no early warning.

1. CURRENT DISMANTLING STRATEGY

The studies leading to shutdown and then dismantling were engaged in 1998 based on a scenario with a status corresponding to IAEA level 2 until 2046.

In 2001, EDF management made the decision to dismantle all the first generation power stations and Creys Malville between now and 2025. It should be noted that the presence of strongly irradiated stellite in the Creys Malville reactor would still require remote systems for working in the reactor block after 2046.

The sequence of operations dictated by the dismantling strategy is as follows:

- Eliminate the risks as soon as possible and in particular the risk related to the sodium, 3300 tonnes of which is kept in liquid form in the reactor vessel;
- Dismantle the most active parts that are too radioactive to be sent to the existing or planned storage centres. This may lead to on-site storage to allow decay to occur before sending to a storage centre;
- Planning of the work interventions in order to limit the costs.

1.1. Sequence of operations

1.1.1. Phase 1

Unloading of the fissile, fertile and absorbent assemblies without installation of dummy assemblies. General dismantling studies and preparation of the files to obtain the decrees necessary for complete dismantling. These files can only be developed once the feasibility studies have been terminated so that a preliminary safety report can be drafted.

1.1.2. Phase 2

- Demolition of the non-nuclear installations;
- Studies and construction of the sodium treatment installation;
- Drilling and siphon installation, construction of the drain line for sodium retentions;
- Removal of the steel assemblies from the lateral neutron shielding;
- Publication of the dismantling decrees;
- First dismantling phase for small primary components;
- Carbonation of the secondary circuits.

1.1.3. Phase 3

- Sodium treatment;
- Dismantling of the secondary circuits;
- Dismantling of the equipment that no longer serves a purpose in the reactor building.

Dismantling of the remainder of the small primary components and all the large components (reactor coolant pumps and intermediate exchangers).

1.1.4. Phase 4

- Inspection at completion of vessel sodium drainage;
- Carbonation of residual sodium in the reactor vessel;
- Washing of carbonate in the vessel.

1.1.5. Phase 5

- Dismantling of the reactor block;
- Cleansing of the concrete in the reactor building.

1.1.6. Phase 6

- Demolition of the reactor building;
- Rehabilitation of the site.

1.2. Main aspects of this strategy

1) Disposal of the caustic soda resulting from the sodium treatment: VLL concrete blocks.

2) Correct operation of the sodium treatment installation: Rhapsody and SDP (Dounreay) experience feedback.

3) Carbonation of the circuits and the vessel (see separate paper). Experience feedback from header and R&D work with the CEA.

4) Feasibility of carbonate washing in the vessel to be demonstrated. This point is closely related to the quality of draining and the effectiveness of carbonation.

DISPOSAL OF CREYS MALVILLE SODIUM

P. ALPHONSE
EDF/CIDEN, D.O. RN, Creys Malville, France

Abstract

When the IAEA dismantling level 1 is reached, the fissile materials have to be removed and the plant fluids and products treated. This document describes the fate decided for the sodium wastes from Superphénix and succinctly describes the method adopted for treating reactor sodium.

1. BACKGROUND - RECAP OF THE CONTEXT (1997/2001)

After the declaration to the French National Assembly on 19 June 1997, of the decision to finally shut down the Creys power plant, and after 6 months of alternating hopes and disappointments as to a possible restart in order to complete the use of the 1st core, and perhaps even the 2nd core, the first structured strategic reflection for definitive shutdown began in February 1998. One of the main aims of this reflection was to prove the technical feasibility of this dismantling based on the assumption of reasonable costs and time limits. In this context of 'technical doubt', fuelled by the media, a very determined attitude had to be adopted.

At the beginning of 1998, it was decided to make a total commitment to this project (fast studies and execution of sodium treatment and dismantling) while at the same time observing the safety and regulatory aspects. This option also met a keen requirement of the Department of the Environment for a technically irreversible operation so that reactor restart would no longer be an option.

The decree of the 31st December 1998 finalized the immediate and definitive shutdown of Creys nuclear power station and authorized unloading of the fuel. It also covers sodium removal and storage operations and the dismantling of the non-nuclear installations within the framework of a safety report and general rules for monitoring and maintenance approved by the DSIN (French Nuclear Installations Safety Authority).

At the EDF-CIDEN, a so-called reference scenario based on the discharge of sodium sulphate salts to the Rhone river (reflecting the Scottish approach used in the sodium disposal process, which is economically less expensive but 'sensitive' in terms of its acceptance by the media and the environmental lobby) has been developed and alternative solutions put forward.

This scenario requires the installation of a primary sodium draining system as well as the treatment of certain significant retentions in the vessel, and the construction of a 'reactor' to transform the sodium into caustic soda. This reactor would be of the same type as on the Sodium Disposal Process for PFR on DOUNREAY site, based on the principle developed by the CEA for the Rapsodie programme (NOAH process). An early commitment to sodium draining before the very specific competences for this complex operation are lost was another strong additional incentive drawn from the experience feedback from other sites.

Faced with the risk relating to the discharge into the Rhone which this scenario involves, a fallback scenario for treatment involving caustic soda cementation and long-term storage was chosen (with several possible technical options). The possibility of intermediate storage of all sodium products, which would release the reactor from the monitoring requirements related to the liquid sodium, was not retained since the cost of this solution would cancel out the indirect gains made on the operating costs.

In 2001, the scenario of sodium sulphate discharge into the Rhone was abandoned in favour of the scenario known as treatment by cementation. In spite of a significant difference in the cost, due to the climate surrounding the Public Inquiry File, it is no longer certain that a new application to discharge to water in this first solution would be looked at favourably or that it would be authorized within reasonable delays. This decision is consolidated by the desire on the part of EDF General Management to proceed to "immediate" dismantling of the nine definitively shutdown nuclear sites, thereby reducing the programming for Creys Malville to a total duration of 25 years. The risk involved in an uncontrollable administrative delay therefore became unacceptable.

To conclude this first part, the choice of the sodium treatment solution for Creys Malville is based on a so-called on-line drainage system (no buffer storage) using tools fabricated for this operation, followed in real time by treatment at 6 metric tonnes/day in two caustic soda transformation reactors (NOAH process) nearly identical to the SDP ones. A total of 13 860 m^3 of caustic soda from primary sodium will be mixed with cement to make packages (26 400 cubes of 1 m^3) that will be stored on the surface since they are only very slightly radioactive (< 100 Bq/g), and this in less than 2 years.

2. TECHNICAL CHOICES FOR PRIMARY SODIUM DRAINING

The pool-type Superphénix reactor (larger version of Phénix) consists of a main vessel filled with 3300 metric tonnes of sodium subjected to only 320 JEPP of radiation and currently maintained at 180°C by an external electric heating system on the security vessel, permissing shut down the primary pumps.

As no clad failure happened the radioactivity of this sodium is relatively low, thereby simplifying the problems of pollution and radiation protection.

The main primary sodium is extracted using an immersed electro-magnetic pump able to output up to 20 metric tonnes/hour into the head tanks upstream of the treatment process (tanks currently store the secondary loop sodium which will have been previously destroyed by the same process). This pump is installed in place of a fuel transfer machine through the upper slab. Special drainage piping is also fabricated for this operation. An alternative gravity run-off solution by drilling through the reactor vessels bottom was examined from the technical and economic standpoints, and rejected.

Prior to this main drainage process, a certain number of additional items of equipment (5 planned to date) will be introduced into the vessel inner structures in order to pump or siphon off the main retention sodium.

An end piece adaptable onto the electromagnetic pump suction line will clean the bottom of the vessel (target: to leave only 40 L maximum). Another end piece will clean the retentions from the double bottom of the recovery plates, themselves drilled prior to draining so that the sodium runs out towards the bottom of the vessel.

Two self-priming siphon systems will eliminate the retentions from the bottom support of the reactor and the connection of the stepped walls and shell located above by gravity flow of the sodium at the bottom of the vessel during draining. Models of these systems are under development or testing with water in order to validate the dimensional and hydraulic data.

All the retentions described above represent a volume of approximately 38 m^3 (1% of the total volume) which once treated will be reduced to 1.4 m^3 (< 0.1% of total volume), except for the residual sodium wetting the surfaces.

Studies to obtain even better results are in hand.

The target is to eliminate as much as possible the sodium after integral carbonation (treatment of approximately 10 to 20 mm sodium thickness).

Sodium draining is accompanied by methods for checking its operation, in particular a periscope, close-circuit TV and bubble-type level gauge installed in the hot sodium manifold.

Means of inspection after draining have also been envisaged. These are probes equipped with ultrasonic sensor or fibroscope to ensure that no retention remains for a good carbonatation and perhaps water filling.

The main identified risks for these operations are the possible escape of sodium (with fire risk — although the sodium is 'low-temperature' which reduces this risk), and the hydrogen risk generated by the transformation into caustic soda. A safety report evaluating these risks in detail and the counter-measures implemented to reduce their occurrence and consequences has been prepared in order to obtain the necessary authorizations from the DSIN.

To conclude, Superphénix vessel sodium draining is entirely carried out using the tools installed on the reactor slab, as envisaged by the design for all the reactor operation equipment. An immersed electromagnetic pump with adjustable flow up to 20 metric tonnes/hour has been manufactured for this operation, which will last less than two years (mid-2007/2009).

Drainage takes place into the head tanks currently used to store the sodium from the secondary loops to be treated in priority (2006/mid-2007), the sodium being transformed into caustic soda then cemented for surface storage of 26 400 m^3 in very low radioactive packages.

SUPERPHÉNIX — STRATEGY AND ORIENTATIONS FOR DISMANTLING THE REACTOR BLOCK

D. CHIAROT
Electricité de France/CIDEN (EDF/CIDEN), Villeurbanne Cedex, France

Abstract

The first studies for dismantling the Superphénix power station reactor block are currently progressing. They have defined the broad outlines of the dismantling strategy and proposed a certain number of orientations. Two dismantling methods have been considered: dismantling with the structures covered with sodium carbonate; and underwater dismantling. 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 parameters contributing to the efficiency of the carbonation operation have been determined in the laboratory. A demonstration on a semi-industrial scale is planned on one sodium loop. The dismantling studies are undertaken with a view to the work lasting eight years maximum.

1. INTRODUCTION

- At present, the dismantling of Superphénix reactor block is planned to begin in 2014 and continue for a period of eight years;
- The preliminary work prior to reactor dismantling will last from mid-2010 until the end of 2013. It will begin after sodium draining from the reactor;
- All the installations should have been dismantled by 2025;
- Figures 1 and 2 show the reactor building and the reactor structures.

2. BACKGROUND

A preliminary approach — reflection in-house at EDF in 1999 — examined three initial conditions for reactor dismantling. These are described below:

- Vessel drained and maintained under inert gas (nitrogen): the internal structures are then either covered with a film of sodium or found under a pool of sodium (called "retention") in the case of non-drainable zones;
- Vessel drained, followed by total carbonation of the internal structures. As this carbonation is never perfect, certain metallic sodium may remain under the thickest retentions;
- Vessel drained and all traces of sodium removed.

In order to check the feasibility of the dismantling operations based on these initial conditions, studies started at the end of 1999 with manufacturers.

The main results of these studies are as follows:

- The installations can be dismantled in the three initial conditions;
- The reactor vessel can be dismantled either from the bottom (from the vessel pit) or from the top (from the slab);
- The most radioactive structures are generally dismantled first;
- The duration of the operations will vary between 6 and 20 years depending on the scenario and the contractors.

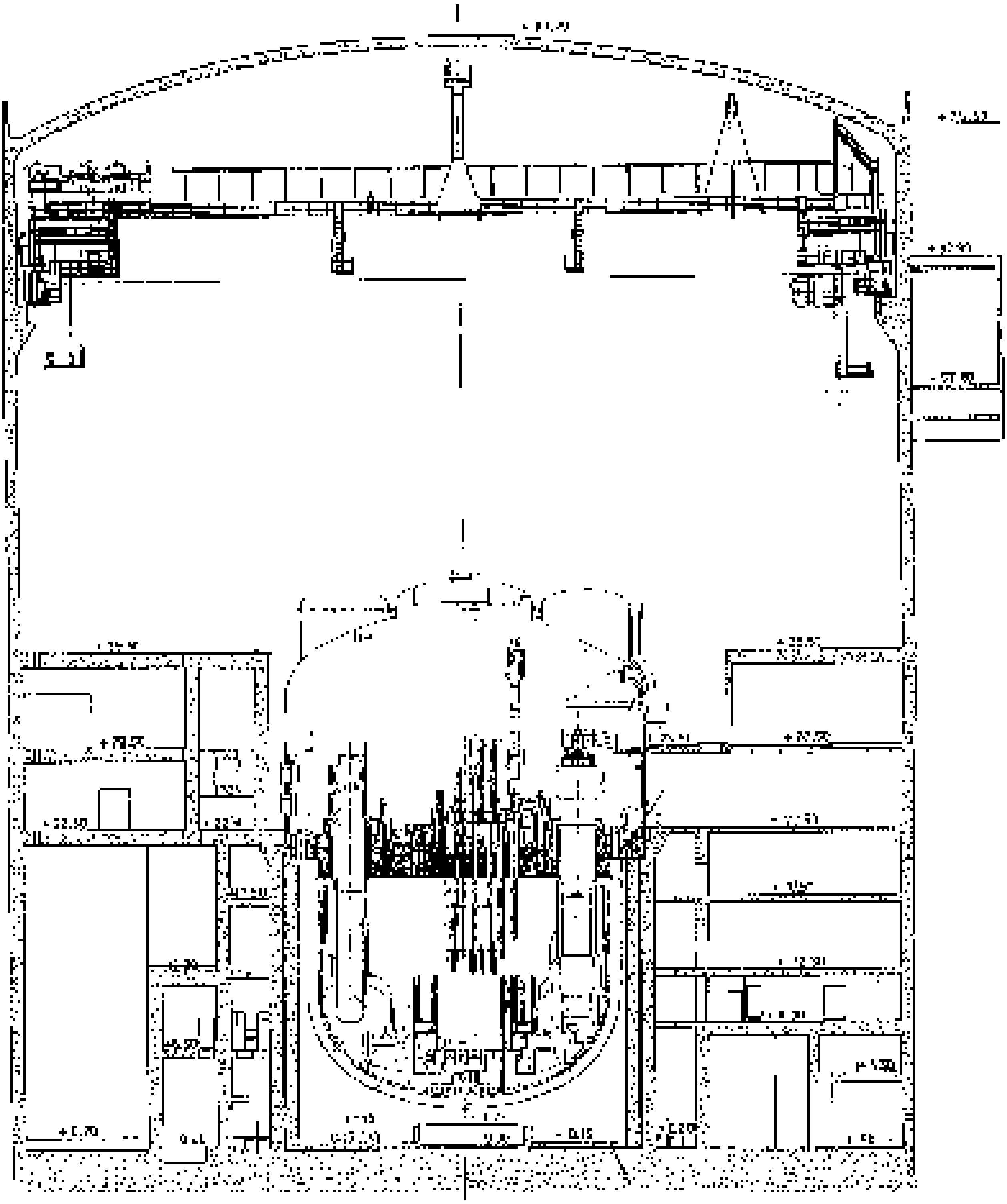


FIG. 1. *Superphénix – Sectional view of the reactor building.*

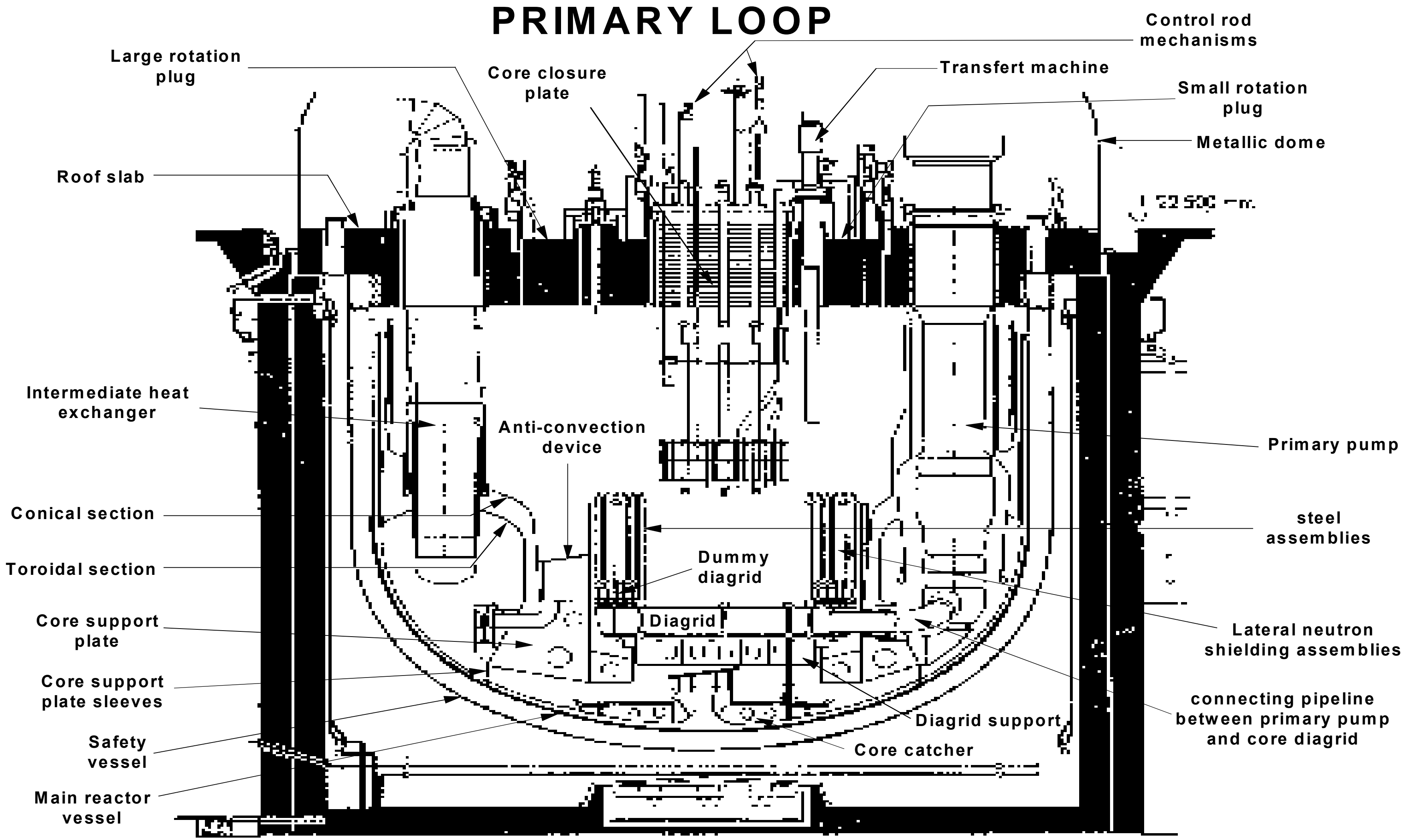


FIG. 2. View of the reactor.

3. STRATEGY

The results of studies led EDF to set down the following principles:

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

3.1. The earliest possible reduction of risks

Although all the dismantling operations generally lead to a reduction of the risk, the period covered by this paper is between the end of reactor block draining (sodium treatment phase) and the end of the reactor dismantling. 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:

- Radioactive sources;
- Metallic sodium.

3.1.1. 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 either 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), or to implement arrangements aimed at assured reductions of the ambient dose rate:

- 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.

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

3.1.2.1. Reduction of metallic sodium related risk

Metallic sodium has 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 liquid 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.

After draining the reactor block, the residual metallic sodium is neutralized by carbonation.

NB: 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.

3.1.2.2. Carbonation of residual sodium

The method 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. Furthermore, the process of carbonation was carried out in 1987 for the dismantling of the intermediate storage vessel for spent fuel in Superphénix.

1) The first point is that the carbonation of the sodium films should not present any particular problem.
2) The second point is that the thickness of the sodium that can be “carbonated in a reasonable time (a few months)” probably has limits: it could be difficult to neutralize the sodium on all the retention thickness if this exceeded 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 at best in the vessel;
- There are few retentions as possible and these are as thin as possible.

An evaluation/confirmation of the achievable performances on a semi-industrial scale (for the reactor block treatment) will be carried out under the TRIPOT test on a sodium loop at the CEA Cadarache. This is a carbonation test on a model representing the discontinuities (retentions, accessibility to carbonation fluid) encountered in the reactor block internal structures. This test could take place during 2004. 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.

3.1.2.3. 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 will give access to certain retentions, it appears convenient to examine the merits of treating them. This reflection will take place while bearing in mind the two dismantling scenarios envisaged. 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.

3.2. The best achievable containment

The state of the installation at the end of the draining operations (fuel removed from the reactor building, primary circuit sodium and secondary loops sodium eliminated), and the presence of small quantities of residual sodium, means that the work will be carried out on the basis of a single containment barrier. However, the principle of precaution will require the installation of an additional temporary containment such as formed by the reactor building enclosure.

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.
For the operations that take place outside the vessel (washing, complementary cutting, etc.), the containment boundary would be situated as close as possible to the structural elements extracted from the reactor. It would be either a containment barrier specific to the structural element (handling cask, vinyl, etc) or the limit of the containment consisting of the waste treatment installation.

3.3. Early dismantling operations to facilitate reactor dismantling

As recapped in Section 3.1.2.3, at the end of 2001, it was proposed to extract and dismantle all the removable components crossing the slab in order to facilitate dismantling of the reactor block. The components will be replaced by plugs:

- Restoring the tightness of the slab containment barrier;
- Ensuring biological shielding of the zone above the slab 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).

3.4. 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 arrangements 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.

4. DECISIONS

The following decisions have been taken based on the strategy defined above.

4.1. Unloading of the steel assemblies

This operation will be carried out before the beginning of reactor dismantling works, and preferably before freezing the sodium left in the vessel after draining. The results of the ideas contest has highlighted the technical difficulties and will extend the dismantling phase due to the presence of these assemblies at the time of reactor dismantling. Moreover, the presence of retentions in the base of these assemblies could exclude underwater dismantling, for reasons of acceptability in terms of safety. Before beginning reactor block dismantling, the steel assemblies will be extracted from the vessel.

4.2. Evacuation of all the removable components through the SLAB

These components will be removed before beginning dismantling the reactor block and before freezing of the sodium in the vessel. The results of the studies has highlighted the technical difficulties and lengthened the dismantling period induced by the presence of these components at the time of reactor dismantling. Before beginning dismantling of the reactor, the removable components penetrating through the slab will have been dismantled (apart from the core cover plug and the rotating plugs). These components will be replaced by plugs restoring the barrier created by the slab and providing biological protection.

4.3. Reduction of the number of dismantling methods

The different dismantling methods envisaged are shown in Fig. 3. The studies by EDF/CIDEN have reduced this number.

4.3.1. Rejection of dismantling operations with structures covered with metallic sodium under inert gas

This method requires dismantling to be pursued under a nitrogen blanket. This requires an airlock for introducing or extracting tools and waste, makes dismantling more complex and increases the risks to personnel. The risks due to the presence of metallic sodium will remain until dismantling of the reactor vessel and its internal structures has been completed. This is a significant risk lasting a considerable time. For this reason, it has been decided to reject this approach. Dismantling from an initial state in which the structures are covered with metallic sodium maintained under an inert gas is rejected.

4.3.2. 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, the arrangements necessary to prepare the future worksite under the vessel and to protect against the spread of contamination should be taken as soon as possible (and as long as the dose rate below the vessel remains reasonable, i.e., before sodium draining).

This method requires gas inerting of the vessel pit in the event of dismantling with metallic sodium. 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. Dismantling via the bottom is rejected. The selected methods of dismantling are shown in Fig. 4.

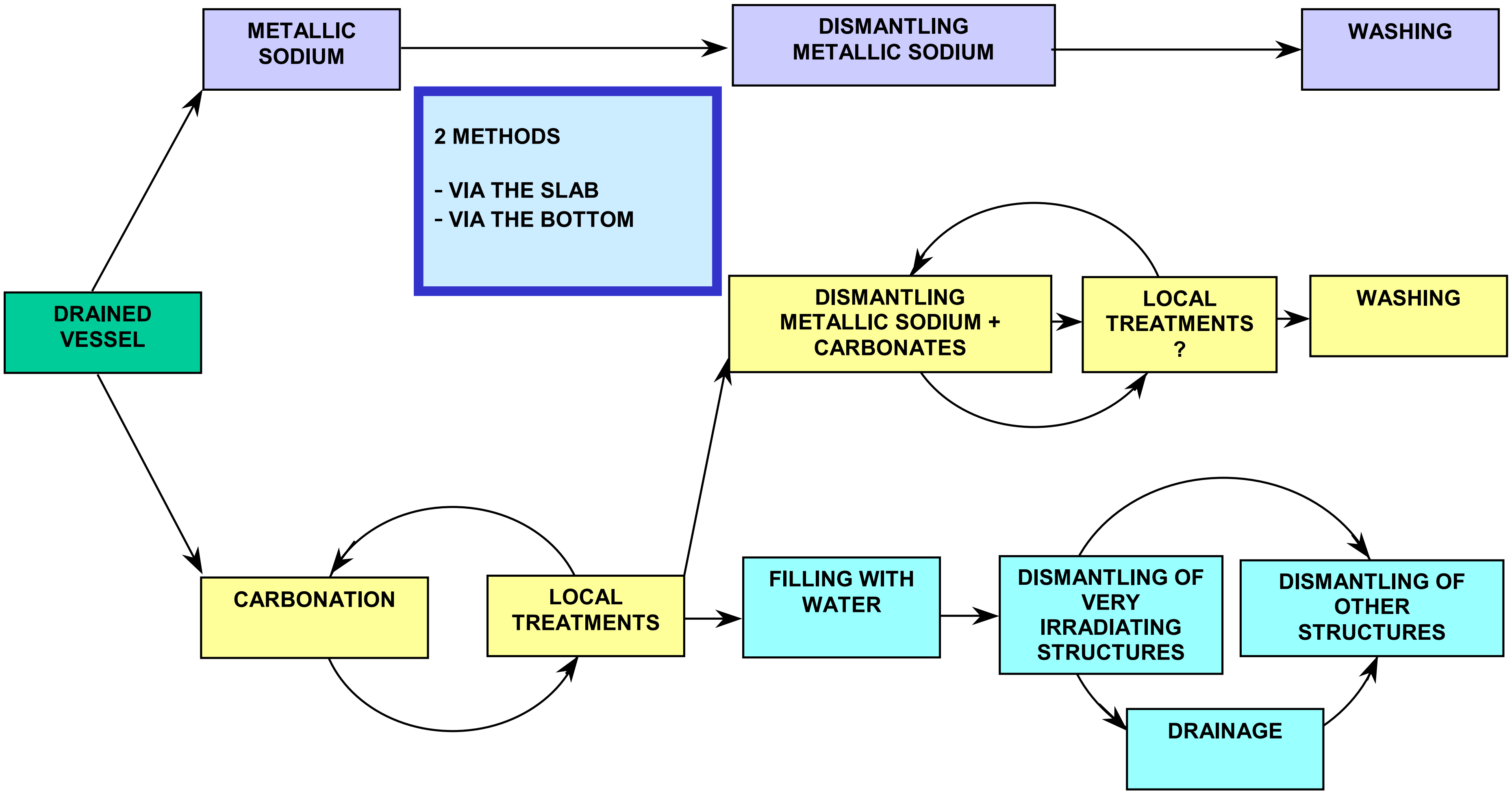


FIG. 3. Dismantling methods envisaged.

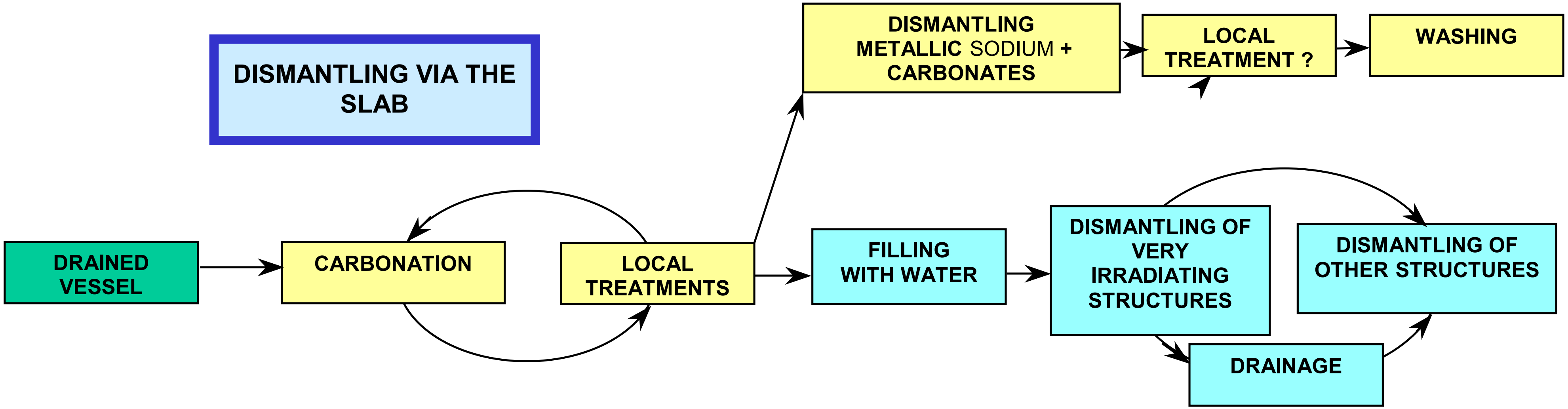


FIG. 4. Dismantling methods selected.

4.4. Operations prior to the two dismantling methods selected

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 that could be as follows:

- Inspection of the vessel internal structures after the draining operation in order to take stock of the condition of the retentions;
- Further treatment of the retentions if required by studies in progress;
- Carbonation (optimized parameters based on preset targets: thorough carbonation, nature of the carbonates required, carbonation time limit, etc);
- Further treatment of the retentions by a chemical method (sodium neutralization by steam and carbon dioxide, isolation of the sodium using a resin to ensure sealing from the ambient conditions, etc) or a mechanical one (scraping, 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 selected, 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.

Dismantling of the reactor will be preceded by a phase for reducing the quantities of metallic sodium in the form of retention followed by a carbonation phase.

4.4.1. 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)	≈ 1.3 m^3
Retentions	≈ 1.3 m^3

The first observation we can make on examination of this table 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 that 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	≈ 0.7 m^3

At present, based on the lessons learned, we are able to guarantee that 10 mm thick of sodium would be neutralized by carbonation (provided that it takes place in the best possible conditions, that is to say, 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 carbonated, at least in laboratory conditions. Therefore, we have set a target which is considered "reasonably achievable", and which is to carbonate 20 mm thick 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 and their location, it seems at present that the underwater dismantling scenario can be applied.

DISMANTLING STRATEGY FOR SECONDARY LOOPS AND ASSOCIATED SODIUM CIRCUITS

E. JOULIA
Electricité de France/CIDEN (EDF/CIDEN), Villeurbanne Cedex, France

Abstract

This document describes the operations to be performed to dismantle the secondary and auxiliary circuits on the Creys-Malville power station. The period covers the disposal of these circuits, from their post-operation status, i.e., secondary sodium drained and circuits maintained under a neutral gas, until complete dismantling of the installations. The problems involved in sodium circuit treatment are more specifically described, and notably the preliminary interventions to materially separate the "secondary" zones from the "primary-reactor" zones and to prepare for the treatment of the residual sodium.

1. GENERAL DESCRIPTION OF THE INSTALLATIONS

The secondary circuit SPX consists of four loops. In each one of these loops, two intermediate exchangers supply a steam generator through two pipes located in the high part of this component. A single pipe (1000 dia.) connects the steam generator return to an expansion tank containing the secondary pump. This feeds the sodium back towards the intermediate exchangers. A sodium-air exchanger erected in parallel allows the reactor residual heat to be evacuated in the event of steam generator unavailability.

Since the power station was shutdown, the secondary sodium has been stored cold in storage tanks located under the steam generators.

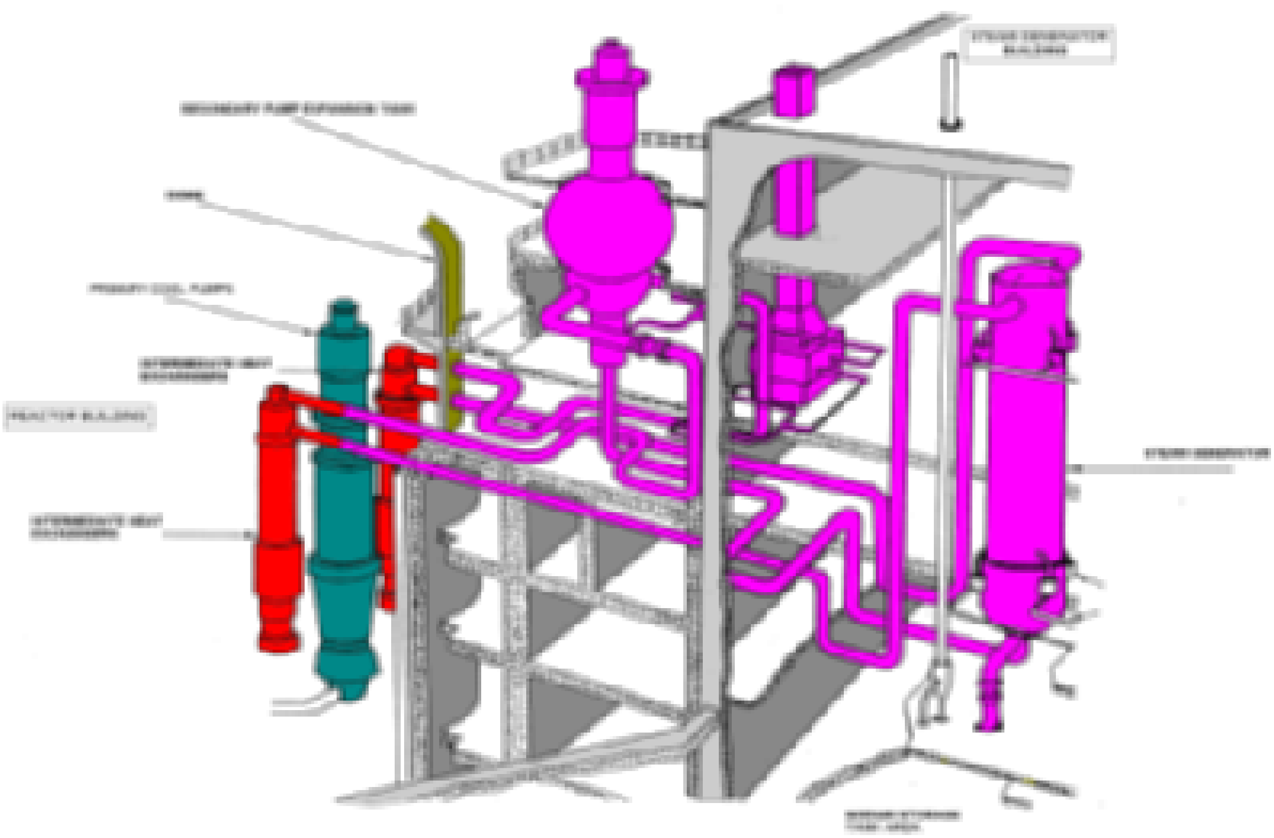


FIG. 1. Main secondary sodium circuits.

Some characteristics of the main secondary sodium circuits:

– Secondary and auxiliary circuits: 4 loops	Length (1 loop): 1200 m Residual Na: 366 kg
– Emergency cooling: 4 loops	Length (1 loop): 2300 m Residual Na: 18 kg

2. DESCRIPTION OF THE PRELIMINARY OPERATIONS

The operations prior to residual sodium treatment and dismantling of the installations are designed to ensure:

- Complementary draining: this draining operation complements the main draining operations carried out at the end of the station operation phase. It is intended to collect together the secondary sodium and the NaK for treatment in the sodium destruction installation. The main operations are as follows:
  - Draining of the integrated auxiliary exchangers and the associated storage tanks;
  - Draining of the intermediate exchangers;
  - Draining of the NaK from the various tanks.

- Main isolations: these isolations are designed to separate the residual heat evacuation systems and the primary circuits materially and functionally, and allow secondary sodium draining from the intermediate exchangers. These isolations will also be used to set up the carbonation installations for treatment of the circuits' residual sodium. The main operations are as follows:
  - Isolation of the intermediate exchangers and the integrated auxiliary exchangers;
  - Isolation of the secondary sodium storage tanks.

3. CARBONATION OPERATIONS

On completion of the preliminary isolation operations and complementary draining, it is planned to treat the residual sodium on the pipework and part of the components by system carbonation.

A system consists of one or more circuits from which the components or discontinuities which are unsuitable for system carbonation treatment are excluded (sodium retentions are too high, pipework clogged by solidified sodium, small pipes or components presenting pressure losses that are too high to allow proper sweeping by the carbonation gas). The components not included in the carbonation system are replaced by flexible devices, temporary pipework or equipment necessary for the carbonation operation (circulation devices and on-line instrumentation) so that the treatment operation can be monitored and selected pipework swept. The network is thus connected to the injection and discharge chambers. The purpose of carbonation is to be able to overcome the risks due to the presence of sodium during the circuit dismantling work. This allows most of the components and pipework to be filled with air. R&D work with the Commissariat à l'Energie Atomique on the sodium carbonation process (Carnac tests) were carried out in 2000 and 2001. These tests have defined:

- The influence of the various parameters: temperature, reaction gas concentrations, etc.;
- The carbonated sodium thicknesses;
- Type of carbonates formed: $NaHCO_3$ or/and Na_2CO_3.

The choice of the carbonation operation is based on experience feedback from secondary pipework treatment operations for inspection and carbonation of the SPX irradiated assemblies storage tank on the Phénix power station at the time of its dismantling. The carbonation of the pipework residual sodium therefore constitutes the reference solution for the treatment of secondary and auxiliary SPX circuits. However, it was decided to engage a demonstration site for treatment of an auxiliary circuit by the carbonation process. The selected auxiliary circuit has been in a so-called "fossil" condition since 1992. Today, this circuit has been partly dismantled and comprises several parts drained and maintained under inert gas. It consists of two identical and independent loops whose main characteristics are as follows:

- Length of one loop: ~270 m;
- Estimated quantity of residual sodium per loop: 12.8 kg;
- Components per loop: 1 circulating pump, 1 heater, 1 clogging indicator, 1 expansion tank, 1 steam trap, 1 sodium/air exchanger, 1 cold trap.

It is planned to proceed to pipework carbonation on one loop and then to dismantling of the installations. The other loop will be dismantled directly without preliminary treatment of the residual sodium. A comparative assessment of the two methods and experience feedback from the operations will be examined. The decision to start a demonstration site for this circuit satisfies the following needs to:

- Study the dismantling operations of an auxiliary sodium circuit on one site by comparing the carbonated and non-carbonated states and performing experience feedback with a view to dismantling all the secondary circuits;
- Materially engage the auxiliary circuits dismantling operations.

4. DISMANTLING OF SG BUILDING INSTALLATIONS - SECONDARY AND ASSOCIATED CIRCUITS

This section concerns the dismantling of all the heat evacuation circuits and other materials contained in the rooms housing the circuits (electric cables, ventilation systems, fire protection systems, lighting, etc.). The significant aspects of SG building installations dismantling operations are:

- Dismantling of the water-steam parts;
- Dismantling of the sodium circuits.

4.1. Dismantling of the water-steam parts

The preliminary stage consists in engaging the dismantling of the "water-steam" rooms of the steam generator buildings in order to have cleared areas for engaging work on dismantling the sodium zones. This work can be started quickly without waiting for work on the sodium zones.

4.2. Dismantling of the sodium circuits

The project data are as follows:

- The residual sodium contained in the circuits has been previously carbonated. However, the presence of residual metal sodium under the layer of carbonate cannot be excluded, on certain components in particular. In situ washing by filling the main piping (1000 and 700 dia.) with water is one option considered;

- The steam generators are system carbonated with the pipework then washed and dismantled in situ;
- Sodium piping is cut up after carbonation and removed to the site treatment workshop for washing.

5. SUMMARY

The scenario for secondary and associated circuit dismantling operations (see Fig. 1) can be divided into three main phases:

1) The preliminary operations of secondary loop isolation from the primary part and residual sodium draining and collection in storage tanks for treatment in the sodium treatment installation. These operations are currently either at the detailed design stage, or pending execution on site: work programmed from 2002 to 2004.

2) The residual sodium treatment of circuits by carbonation. The demonstration site should make it possible to decide on the merits of this type of treatment on comparable circuits. Even so, preliminary carbonation of the secondary/main circuits with specific components such as the SGs seems unavoidable. Preliminary studies of the carbonation of these circuits have been performed. The pilot site is planned for the beginning of 2003 and the application to the other secondary circuits from 2004 to 2006.

3) The dismantling operations on the heat evacuation circuits (secondary and auxiliary circuits).

The reference scenario retained is as follows:

- Treatment of the “water-steam” rooms and the sodium-free parts is engaged in priority;
- The SGs are system carbonated with the pipework before being washed and dismantled in-situ;
- The sodium pipework is cut up and removed for washing in a workshop for treatment of secondary components located on the site. The main piping (1000 and 700 dia.) could be washed in situ before cutting operations.

Work planned from 2005 to 2010.

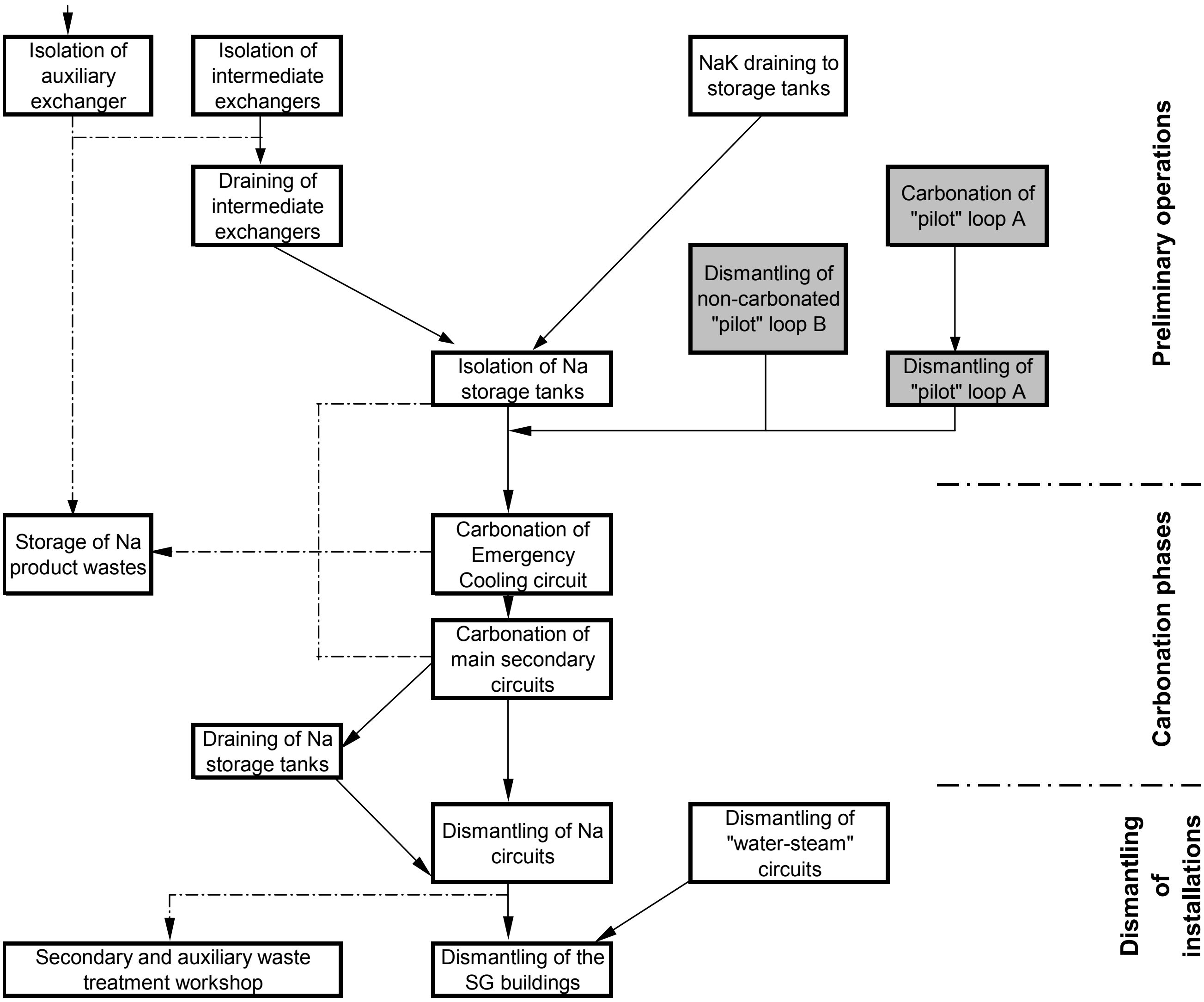


FIG. 1. Flow sheet for the dismantling of the secondary sodium loops and the associated sodium circuits.

CONCEPT FOR DISMANTLING THE REACTOR VESSEL AND THE BIOLOGICAL SHIELD OF THE COMPACT SODIUM COOLED NUCLEAR REACTOR FACILITY (KNK)

I. HILLEBRAND, W. PFEIFER
Forschungszentrum Karlsruhe, Karlsruhe, Germany

Abstract

The Compact Sodium-cooled Nuclear Reactor Facility (KNK) was an experimental nuclear power plant of 20 MW electric power erected on the premises of the Karlsruhe Research Center. The plant was initially run as KNK I with a thermal core between 1971 and 1974 and then, between 1977 and 1991, with a fast core as the KNK II fast breeder plant. Under the decommissioning concept, the plant is to be decommissioned completely to green field conditions at the end of 2005 in ten steps, i.e. under the corresponding ten decommissioning permits. To this day, nine decommissioning permits have been issued, the first one in 1993 and the most recent one, number nine, in 2001. The decommissioning and demolition activities covered by decommissioning permits 1 to 7 have been completed. Under the 8th Decommissioning Permit, the components of the primary system and the rotating reactor top shield are to be removed by late 2001. Under the 9th Decommissioning Permit, the reactor vessel with its internals, the primary shield, and the biological shield are to be dismantled. The residual sodium volume in the reactor vessel was estimated to amount to approx. 30 L. The maximum Co-60 activation is on the order of 10^7 - 10^8 Bq/g; the maximum dose rate in the middle of the vessel was measured in April 1997 to be 55 Sv/h. The difficulty involved especially in dismantling KNK, on the one hand, is posed by the residual sodium in the plant, which determines the choice of neither wet nor thermical techniques to be used in disassembly. Another difficulty is caused by the depth of activation by fast neutrons, as a result of which not only the reactor vessel proper, but also the entire primary shield (60 cm of grey cast iron) and large parts of the biological shield must be disassembled and disposed of under remote control.

1. INTRODUCTION

The Compact Sodium-cooled Nuclear Reactor Facility (KNK) was an experimental nuclear power plant of 20 MW electric power erected on the premises of the Karlsruhe Research Center. The plant was initially run as KNK I with a thermal core between 1971 and 1974 and then, between 1977 and 1991, with a fast core as the KNK II fast breeder plant.

The reactor core of KNK was arranged in an unpressurized, thin-walled reactor vessel roughly in the middle of the containment (Fig. 1). Sodium was used as the coolant.

The entire nuclear fuel and all movable core internals have already been disposed of. The sodium coolant has been removed except for some residues clinging to inner surfaces and in inaccessible locations. The tertiary systems (water-steam loop with the turbine) and the secondary sodium systems, including the associated auxiliary systems and buildings, have been taken out and demolished, respectively. The ventilation system, the electricity supply facilities, and the reactor entrance and exit lock have been adapted to the requirements of the decommissioning steps to follow. The primary system including the primary sodium dump tank and the fuel element store and the rotating reactor top shield of the reactor vessel were dismantled. The work conducted so far has been based on eight decommissioning permits.

All activated and/or contaminated materials are transferred to the Central Decontamination Department (HDB) of the Karlsruhe Research Center, which processes them in line with its permit under the Atomic Energy Act, and holds them in temporary storage.

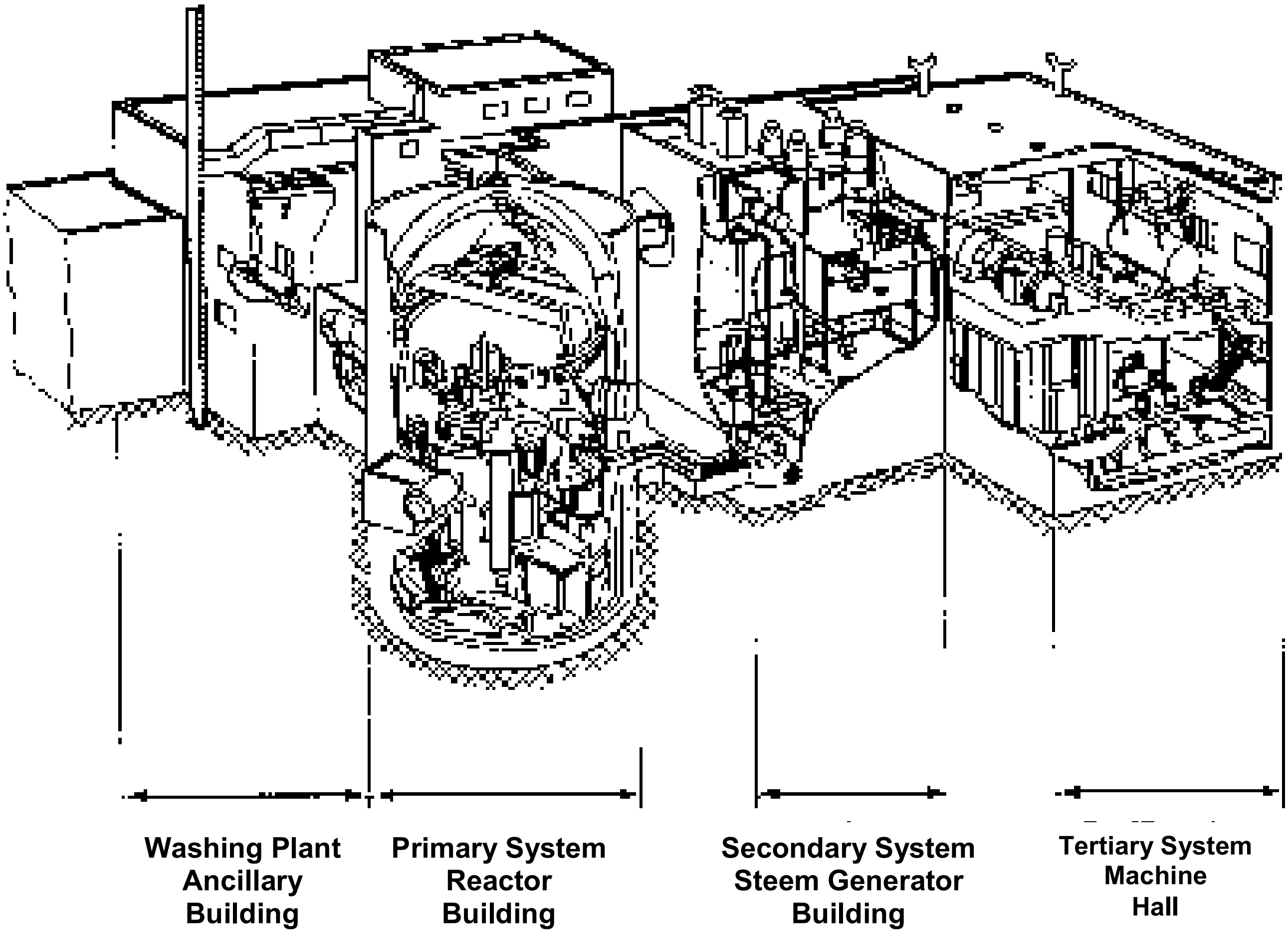


FIG. 1. KNK systems and buildings.

Under the 9th Decommissioning Permit, the reactor vessel with its internals, the primary shield, and the biological shield are to be dismantled. A Europeanwide limited tendering procedure was first run for these activities, and at last the contract was made with Westinghouse Reaktor Germany.

The difficulty involved especially in dismantling KNK, on the one hand, is posed by the residual sodium in the plant. This determines the choice of techniques to be used in disassembly and, in addition, the material must either be removed or converted by chemical means after component disassembly, as components bearing sodium metal cannot be delivered to HDB or stored in a repository.

Another difficulty is caused by the depth of activation by fast neutrons, as a result of which not only the reactor vessel proper, but also the entire primary shield (60 cm of grey cast iron) and large parts of the biological shield must be disassembled and disposed of under remote control.

2. PERMITS AND DEADLINES

Under the decommissioning concept, the plant is to be decommissioned completely to green field conditions in ten steps, i.e. under the corresponding ten decommissioning permits. To this day, nine decommissioning permits have been issued, the first one in 1993 and the most recent one, number nine, in 2001.

The decommissioning and demolition activities covered by decommissioning permits 1 to 7 have been completed. Under the 8th Decommissioning Permit, the components of the primary system and the rotating reactor top shield are to be removed by late 2001.

The 9th Decommissioning Permit covering disassembly of the reactor vessel and the biological shield was filed for in July 1999 and, with a final amendment, again in March 2000. The expert opinion covering these activities has been available since December 2000, and the permit was issued in March 2001. The period between 2001 and mid-2002 has been reserved for planning and preparing disassembly of the reactor vessel. From September 2002 on, the reactor vessel with its internals is to be disassembled and disposed of, and from mid-2003 on, the primary shield and the biological shield are to be disassembled and disposed of.

Under the 10th and last Decommissioning Permit, the remaining auxiliary systems (sodium washing plant, ventilation plant, liquid effluent system, gaseous effluent system, etc.) are to be dismantled and any buildings remaining are to be decontaminated, measured for clearance, and then demolished, if necessary. Then the site is to be recultivated.

The safety report on which the application for the 10th Decommissioning Permit is based is being completed and was submitted to the authority in June 2001. The work is to be finished probably at the end of 2005.

3. DISMANTLING THE REACTOR VESSEL AND THE BIOLOGICAL SHIELD

3.1. Initial condition

After completion of the first eight decommissioning permits, the only remnants of the plant still in existence are the reactor vessel with its internals installed in the primary shield and the biological shield. These components are located in the middle of the containment in the reactor building (see Fig. 2). The reactor vessel is inerted with nitrogen and closed with a lid. Other installations still in place are the ancillary plants building, the control room building, and a storage facility. They contain some systems important in the decommissioning process, namely the ventilation system, the washing system for components wetted with sodium, and the moderator store, which must be converted into a buffer store. The reactor building and the ancillary plants building are part of the controlled area.

The residual sodium volume in the reactor vessel was estimated to amount to approx. 30 L. The maximum Co-60 activation is on the order of 10^7 - 10^8 Bq/g; the maximum dose rate in the middle of the vessel was measured in April 1997 to be 55 Sv/h.

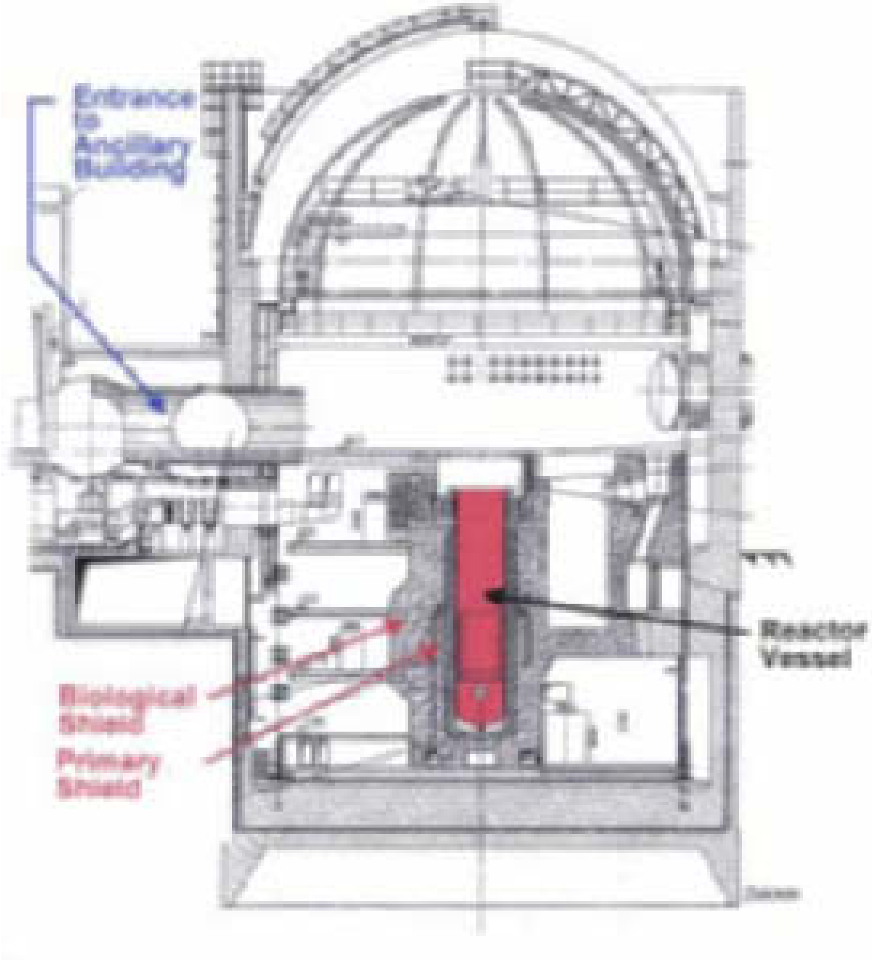


FIG. 2. Cross-section through the KNK containment after completion of the 8th decommissioning permit.

3.2. Demolition concept

Before dismantling of the reactor vessel is begun, the interior space of the vessel is treated with a wet gas. For this purpose, the nitrogen is added humidity so that any particulate sodium deposits can be immobilized. Then the vessel is dried. The dismantling work of possibly sodium covered components is to be carried out under a nitrogen atmosphere.

The reactor vessel with its internals, and the primary shield, are to be disassembled within the existing shielding, i.e. the biological shield. For this purpose, a shielding enclosure will be erected at runway level above the reactor vessel, which will be equipped with a handling cell, an intervention cell, a double-lid lock and a transfer lock for building rubbish, and all the necessary auxiliary systems (lifting gear, rails, lead glass windows, manipulators). The enclosure must have a shielding of 35 cm of steel required for radiological reasons, on the one hand, and ensure separation from the containment in terms of ventilation, on the other hand.

Because of the hazard of sodium fires, only mechanical cutting techniques, such as sawing, milling, drilling, or cutting, may be used to dismantle the reactor vessel and its internals. Abrasive cutting and thermal cutting techniques may be used only with components free from sodium, such as the primary shield, or the reinforcement in the biological shield. The machines required for dismantling are to be mounted on a carrier that can be positioned variably inside the vessel and the biological shield, respectively. The necessary support systems and devices/auxiliary tools are to be harmonized and, as a consequence, minimized in number.

To minimize the exposure dose to the personnel disassembling these systems, and to minimize the number of transports, the radioactive components to be demolished will not be moved to HDB in larger sections. The packages to be used for nearly all metal components are 150 L drums or, for components wetted with sodium, the corresponding 150 L baskets, which will be packed in 200 L drums through a double-lid system and placed in shielded casks for transport to HDB. The components wetted with sodium must be transported first to the washing plant in a shielding bell to be cleared of sodium before they can be packed in 200 L drums. Shielded type-II KONRAD containers will be used for the concrete rubbish. The number of packages produced is to be optimized in order to save costs of interim storage and final storage.

3.3. Disassembly of the reactor vessel

Table 1 provides data about the geometry, mass, and activity of components.

TABLE 1. GEOMETRY, MASS, AND ACTIVITY OF COMPONENTS

Component	Height (mm)	Thickness/ diameter (mm)	Mass (mg)	Max. activation on 1st January 2001 (Bq/g)
Reflector	2310	70-170	11.8	3.1 E+7
Thermal shield	2310	80	7.8	4.8 E+6
Thermal shock liner	6500	12	3.8	4.2 E+6
Other internals	-	-	2.8	1.2 E+9
Internal vessel	10500	16	11.8	4.0 E+6
External vessel	9500	12	4.8	2.2 E+6
Total			42.8	

As a rule, the activation was calculated on the basis of a cobalt content of the steel of 200 ppm. The stellite bushings of the cladding tube plate, have a much more high cobalt content, so they show a maximum activation of 10^9 Bq/g.

The internals of the reactor vessel are to be demolished inside out. The internal vessel and the external vessel must be demolished from bottom to top because they are suspended from an upper flange. The design of the reactor vessel can be seen from Fig. 3.

A central mast manipulator is to be introduced into the reactor vessel for disassembly purposes; it can be positioned in a variety of locations and will achieve self-bracing at the level of the cutting position. The manipulator must be designed so that it can handle, by means of a carrier system, all tools needed to disassemble all internals and the vessel proper. All cuts must be made so that the parts can be packaged in 150 L drums or baskets. In disassembly, special attention must be paid to the cladding tube plate with the stellite bushings and to the double-walled pipe joints cut out of the reactor vessel.

After disassembly of the metal components, the thermal insulation made of fireclay around the reactor vessel must be removed. This can be achieved either by cutting, as mentioned above, or by chipping, as in the later demolition of the biological shield.

3.3. Disassembly of the primary shield

At the level of the reactor core, the primary shield made of cast iron with lamellar graphite, GG-20, is situated in a niche of the biological shield outside the thermal insulation (see Fig. 3).

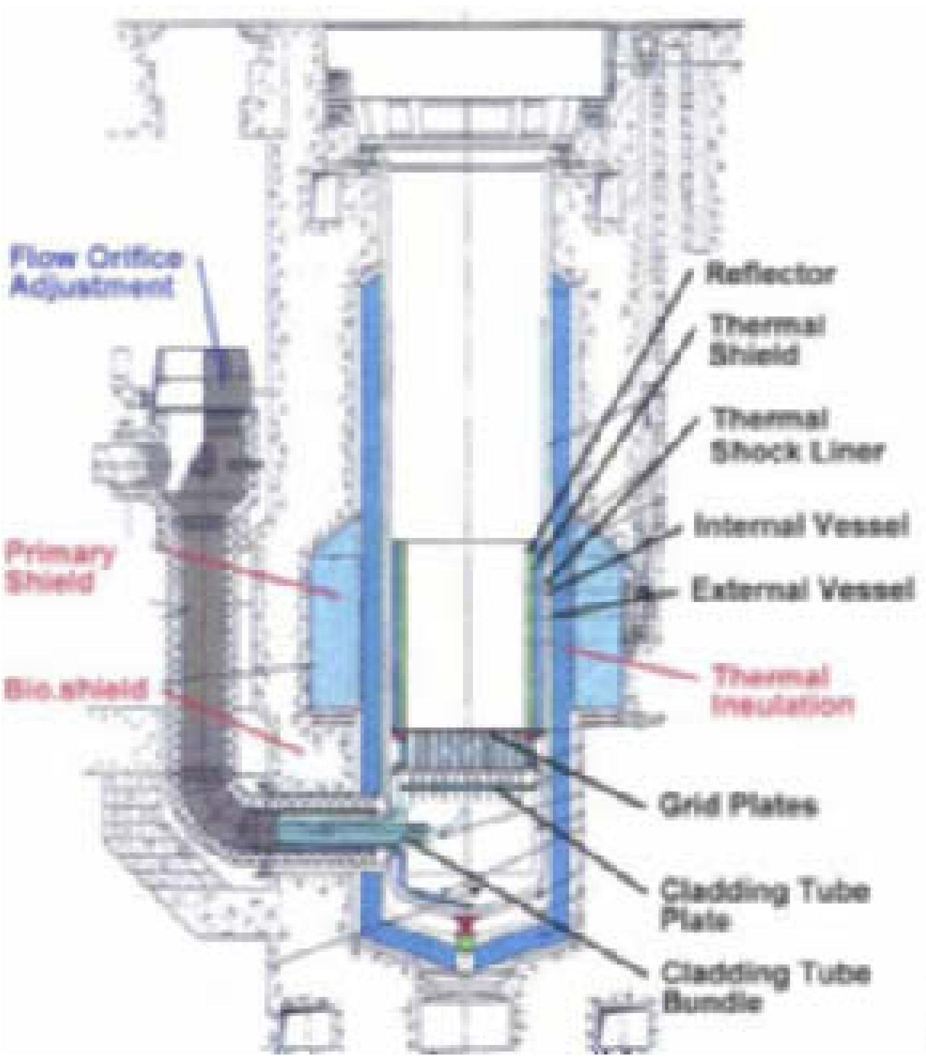


FIG. 3. Reactor vessel with internals.

The total mass of the primary shield is approx. 90.5 mg. Of this, 9.2 mg is due to the conical part, whose four segments are approximately equal in weight. The triangular segments of the cylindrical section each have a mass of approx. 3.7 mg, while the square ones have a mass of approx. 16.6 mg. The maximum activation is 1.55×10^6 Bq/g.

Perhaps it will be necessary, prior to demolition of the primary shield, to remove parts of the biological shield above the primary shield as far as the outside diameter of the primary shield. This makes the primary shield freely accessible from the top and from the inside.

The primary shield is demolished by remote operation from the enclosure. The planning is to install a platform below the primary shield in the reactor cavity on which parts of the primary shield can be deposited. The parts of the primary shield are then disassembled by means of a saw which can be carried by the crane; it is applied to the component, braced, and thus allows horizontal and vertical cutting.

The parts cut off are attached to the crane by means of force-locking mechanical grabs, and are lifted to the runway level. The parts are filled into 200 L drums through the double-lid lock, and are then taken to HDB in shielded shipping casks.

3.5. Disassembly of the biological shield

The reactor core is surrounded by a block of concrete of very high density (density 4.14 g/cm^3), the biological shield, which was also activated by the neutron radiation emanating from the reactor core (see Fig. 3). The specific Co-60 activity of the concrete achieves a maximum of 8×10^5 Bq/g, which means that most of the disassembly work must be carried out remotely.

The depth of demolition of the biological shield is determined by the depth of activation of the concrete. According to the new German Radiation Protection Ordinance, a clearance level for Co-60 of 0.09 Bq/g must be observed for the unrestricted clearance of building rubbish. Probably, a total of 330 mg of very-high-density concrete must be disposed of as radioactive waste.

Chipping will be the method of demolition (Fig. 4). For this purpose, an auxiliary platform variable in height is to be positioned in the reactor cavity, on which a small excavator will be placed. The concrete rubbish is to be sucked through a hose right into the type-II KONRAD container set up at the building rubbish transfer lock of the containment.

The reinforcement bars in the concrete must be cut mechanically or thermally at the same time and disposed of in 200 L drums through the double-lid lock.

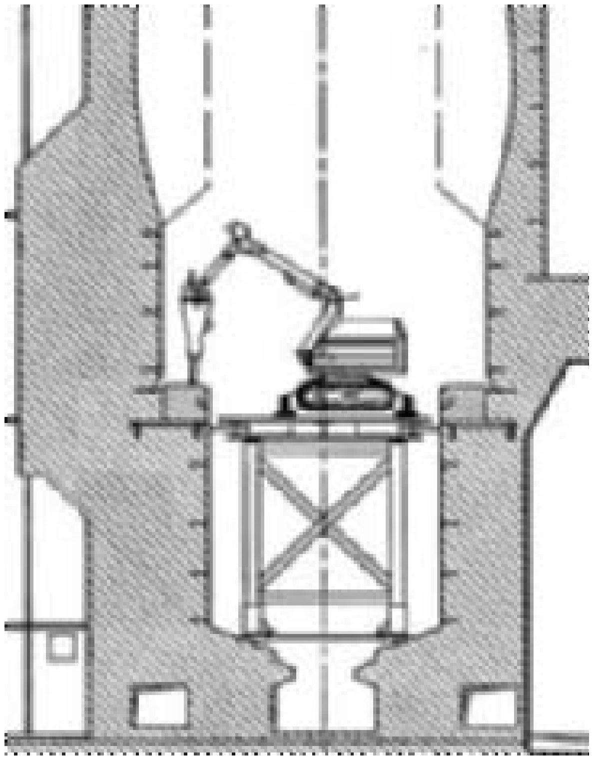


FIG. 4. Demolition of the biological shield.

Table 2 provides the data about the balance of the radioactive waste.

TABLE 2. BALANCE OF THE RADIOACTIVE WASTE

Type of residue	Mass (mg)	Max. activation on 1 January 2001 (Bq/g)	Total activity Co-60 (Bq)
Steel	approx. 43	3 E+07	1,7 E+14
Fireclay	approx. 28	9 E+04	-
Grey cast iron	approx. 90	1 E+06	2,5 E+13
Very-high-density concrete	approx. 330	7 E+05	6.2 E+12

The residues steel and grey cast iron will be packaged into 200 L-drums and stored in the HDB interim storage for ILW-Waste. After some half-life times of cobalt and packaging the drums into a shielded container, the waste fulfill the KONRAD repository requirements.

The residues fireclay and concrete will be directly packaged into shielded KONRAD containers and stored in the HDB interim storage for LLW-Waste, ready for final disposal.

EXPERIMENTAL FEEDBACK ON SODIUM LOOP DECOMMISSIONING

O. VINOCHE, G. RODRIGUEZ
Commissariat a l'Energie Atomique (CEA), Saint-Paul-Lez Durance Cedex, France

Abstract

The aim of this paper is to present experimental feedback on sodium loop dismantling techniques at the CEA (The French Atomic Energy Commission) and to offer recommendations for the decommissioning of Fast Reactor secondary sodium loops. This study is based on acquired CEA decommissioning experience which primarily concerns the following: the decommissioning of RAPSODIE (France's first Fast Reactor), the Phénix reactor secondary loop replacement, the sodium loop decommissioning carried out by the Laboratory of Sodium Technologies and Treatment, and several technical documents. This paper deals with the main results of this survey. First, a comparison of 8 pipe-cutting techniques is made, taking into account speed in cutting, reliability, dissemination, fire risk due to the presence of sodium, cutting depth, and different types of waste (empty pipes, sodium-filled pipes, tanks...). This comparison has led us to recommend the use of an alternative saw or a chain saw rather than the use of the plasma torch or grinder. Different techniques are recommended depending on if they are on-site, initial cuttings or if they are to be carried out in a specially-designed facility referred to hereafter as "the cutting building". After the cutting stage, the sodium waste must be processed with water to become an ultimate stable waste. Four treatment processes are compared with different standards: speed, cost, low activity adaptability and "large sodium quantity" adaptability. Recommendations are also made for reliable storage, and for the general dismantling system organization. Last, calculations are presented concerning a complete dismantling facility prototype capable of treating large amounts and volume of sodium wastes.

1. INTRODUCTION

Sodium is primarily used in the nuclear industry as a coolant in primary and secondary loops of Liquid Metal Fast Reactors. Several LMFRs will be shut down in coming years, and thus many sodium loops will have to be decommissioned. Metal sodium is forbidden in both active and inactive waste for storage because it cannot be considered as ultimate waste: sodium reacts when coming into contact with water and air, and therefore must be processed to become a chemically stable product. In order to accomplish this transformation, a specific process must be applied using a reagent. In most cases, this reagent is water. All waste components containing sodium must be cut into reduced parts, and disposed in such a way that the sodium itself can be treated. All these operations have been conducted at the CEA on several LMFR loops including RAPSODIE (the first LMFR in France) and the PHÉNIX reactor but also on some experimental loops. The aim of this document is to present and assess the experimental feedback in this field in order to establish the general organization of a sodium loop decommissioning system, and to determine what are the most efficient techniques to be implemented when dealing with the very specific characteristics of sodium.

2. GENERAL ORGANIZATION SYSTEM [1, 2]

2.1. Preliminary stages

1) Emptying of the loop;
2) Estimation of the residual mass of sodium (inspection, calculation…);
3) Electricity dismantling;
4) Removal of the insulating material/instruments.

Long period heated insulating material can become pulverulent. In that case, insulating material and iron sheet should be dismantled in the same time. The system is then isolated from premises and ventilated. Insulating material is wrapped into vinyl.

2.2. System main stages

On–site cutting

↓

First storage

↓

Cutting in specified building

↓

Second storage

↓

Processing installation

↓

Final storage

A dismantling procedure is written before each dismantling system is established. It describes the dismantling chronology and the steps of the system.

First, the cutting lines are drawn on the loop with the maximal precision, or marked on a plan. Pipes are attached to the existent frame by temporary hoists to avoid falls during the cutting operations. As in the majority of the dismantling operations, the risk of a fall by either the equipment or a worker is high.

Experimental feedback from CEA sodium loop dismantling operations has clearly demonstrated that this is the major risk, far beyond that of the hazards involved in handling the sodium.

3. CUTTING STAGE [3]

Pipe and component cutting is necessary to allow a thorough treatment of the residual sodium which is carried out according to a special process. The number of cuts to be made, and the geometrical configuration of the waste are determined by the treatment process which is selected much sooner. In every large-scale decommissioning work site studied, the cutting stage was divided into two phases: an initial, on-site cutting was carried out on the loop using portable tools in order to extract long pipes and components. Then, the second cutting phase was conducted inside a specially-designed “cutting building”. This type of organization presents a major advantage because it allows workers to handle large quantities of waste in a specifically-designed structure, and not next to the loop itself where space and the maneuverability of handling tools are limited. Components are systematically moved to the cutting building except tanks that are cut on-site into reduced parts. Only long parts of pipes are cut on-site, and entry and exit component pipes. Four meters is the average length of the pipe parts that are removed and transferred to the cutting building.

FIG. 1a. General view of the loop before operations.

FIG. 1b. Removal of the insulating material.

FIG. 1c. Removal of the instruments.

FIG. 1d. Reciprocating saw cutting.

FIG. 1e. General view after decommissioning.

FIG. 1(a-e) Several pictures of a dismantling system (EPINAR test loop).

This average depends on handling tools and accessibility. The cutting stage can be tricky when dealing with the variable pipe lengths and diameters, and with the diversity of the components (pumps, gauges, valves, cold traps…). Different tools may be used but they must meet several criteries: reliability, speed, cutting depth, low secondary emission, reduced sodium heat risk. Lubrification must be as low as possible to avoid secondary effluent emission. Carbonation is recommended for large-scale loops, in order limit the sodium risk. The small size of the loops at the CEA does not justify using this technique. Sodium loop cutting can be divided into successive steps. Components and pipes are bound to avoid falls, vibrations during cutting and saw blade jamming. Cutting starts with the upper parts of the loops in order to limit falls.

Metallic chips resulting from the sawing must be retrieved. These chips are to be classified in the same category as sodium waste and they too must be processed. Components with a complex internal structure (e.g. cold traps) are not drained if possible, because solid sodium keeps the internal structure in place and limits vibration and overheating during cutting operations. The main cutting techniques used in the industry have been compared with a relative notation system for each criteria from 0 to 5. Estimations have been realized for missing data. Results for each techniques have then been compared for different waste families (small size massive components, variable diameters pipes, tanks).

These techniques are the following:

- Reciprocating saw;
- Band saw;
- Grinder;
- Plasma torch;
- Circular saw;
- Hydraulic shears;
- Nibbling machine;
- Pipe cutter.

This comparison reveals that mechanical cutting techniques are preferable. Thermal techniques are faster for stainless steel cutting but slower for sodium cutting. Moreover, thermal techniques heat up the sodium (a neutral gas is then necessary to avoid burning) and present a dissemination risk when dealing with active sodium. The main recommended cutting techniques for sodium decommissioning loop system is set down in the following tables. These results take into account all the criteria studied.

3.1. On-site cutting

TABLE 1. RESULTS FOR ON-SITE CUTTING

Component family	Most efficient technique	Breakdown tools
Low diameter pipe (DN 25 maxi)	Portable band saw	Reciprocating portable saw Hydraulic shears Pipe cutter
Cylindrical component (no internal structure) (DN 500 maxi)	Portable band saw	Nibbling machine Reciprocating portable saw Hydraulic shears Pipe cutter
High diameter pipes (DN 2200 maxi)	Circular saw	Nibbling machine Pipe cutter

TABLE 2. CUTTING IN A SPECIFIED BUILDING

Components family	Most efficient technique	Complementary technique
Small size massive components (200 × 200 maxi)	Band saw	Reciprocating saw
Low diameter pipe (DN 25 maxi)		Reciprocating saw Nibbling machine
Cylindrical component (no internal structure) (Ø 500 mm maxi)		Reciprocating saw Nibbling machine
Cylindrical component (internal structure) (Ø 1500 maxi)		Reciprocating saw Circular saw + diamond cable
High diameter pipes (Ø 2200 maxi)		Nibbling machine Reciprocating saw Circular saw

4. TEMPORARY STORAGE

Storage before cutting in a specified building and storage before treatment should be limited, if achievable, to one workday maximum so that the systematic use of a waste container can be avoided (neutral gas is necessary). This storage can be safe at ambient air temperature providing the humidity is restricted to limit reactions at 2.5 grams of water for 1 kg of air.

During on-site cutting, the pipe parts are isolated with vinyl and tape, or with soldered steel. Pipe parts can thereby be moved quite safely to the cutting building. Metallic containers filled with neutral gas should be available for potential longer storage (more than one week) or for a halt of the works. In case of long storage, the neutral gas of the containers is renewed every 3 months.

5. SODIUM TREATMENT PROCESSES [5, 6]

The purpose of the process consists in turning sodium into a stable product vis à vis water and air. This process must be efficient, adjustable (it must be applicable to all sodium loop constituents), reliable and sufficiently quick. Moreover, researchers must be able to check effluent composition and activity.

Processes approved for sodium treatment are water chemical methods. Alcohol is forbidden by the CEA since an accident occurred during the decommissioning of the RAPSODIE reactor in 1994 [7, 8]. Physical cleaning methods (e.g. scraping, evaporation) are more limited at an industrial scale due to their relatively tedious nature.

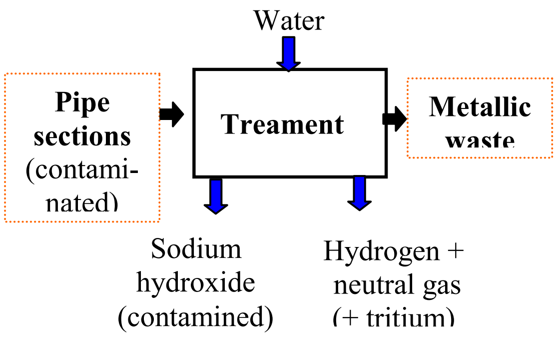


FIG. 2. Water treatment process diagram.

Solid sodium treatment with water is carried out according to the reaction:

$$Na_{solid} + H_2O_{liquid} = NaOH \quad + 1/2\, H_{2\,gas} + 138\ kJ$$

25°C 25°C anhydrid 25°C exothrmic

cristallised

$$NaOH_{anhydrid} + H_2O_{excss} = NaOH, H_2O_{aqueous\ solution}$$

With an excess of water a solution of sodium hydroxide is produced. This solution can then be transformed into a chemically stable and solid waste, by cementation or ceramization process. Hydrogen must be dehumidified and evacuated after radioactive control in case of active sodium.

Water can be projected as a liquid jet, as fog, or as vapor. Treatment speed differs according to the selected process but the chemical reaction and the effluents are the same (sodium hydroxide and hydrogen). The main precaution to be taken to prevent hydrogen risk is to work under neutral gas. The explosive O_2-H_2 mixture is thus avoided. In the case of the process under air, a special study should be carried out to assess the capacity of an armored structure in withstanding a fixed and known explosive volume. A mastery of the process is achieved by acting on one of the two reagents:

- By controlling and limiting water flow (limited by the operator's action or by the process regulation);
- By controlling and limiting the sodium quantity to be processed (restricted by the dimensions of the processing facility).

Processes have been compared with the following criteria: speed in processing, cost effectiveness, contaminated sodium adaptability, waste configuration changes in the baskets, large quantity application, effluent volume production. The processes compared are the following:

— Cleaning pits;
— Autoclave reactor;
— Water jet treatment;
— Active cleaning enclosure;
— Results of the comparison.

5.1. Cleaning pits

They are used in usual maintenance operations or for final treatment of contaminated sodium components. The treatment is slow, to allow a soft sodium/water reaction, in order to re-use or to assess the component. Generally, reaction is obtained by a water flow under a neutral gas or under carbon dioxide for sodium hydroxide carbonation. These processes can be used for LMFR decommissioning operations; the cleaning pits are built in every LMFR. The drawbacks are the sodium acceptability that can be limited to small quantities, the place where they are built (waste handling and managing), their working schedule. For these reasons, the cleaning pits are not optimal for the majority of sodium loop dismantling systems.

5.2. Autoclave reactor

This process is designed to treat small quantities of sodium with an excess of water and to support the rapid raise of pressure due to the reaction. Sodium is placed in a vessel in the upper part of the autoclave. Inside air is swept out and the reactor is filled with a neutral gas. The bottom of the reactor is then filled up with water. Sodium is dropped into this water volume, and a violent sodium/water reaction is obtained. When the pressure is stabilized, reactor is decompressed and gas effluents are diluted after filtration and dehumidification.

This process presents the advantages of being safe and efficient. Moreover, effluents are easily controllable. Its use is optimal for small quantities of sodium (less than 1 kg per load), but it is not efficient for the treatment of parts of sodium loops.

FIG. 3. View of the autoclave installation.

5.3. Water jet treatment

Sodium wastes are processed by a discontinue water jet, thrown by a hose, under air atmosphere. Operators are outside the process area and they drive the hose. Sodium/water reactions are sudden. The main advantage of this process is its processing speed, and its adaptability. Though, gas effluents control is tricky.

This process, mainly used on CEA loops decommissioning, has proven its reliability and efficiency on inactive sodium.

FIG. 4 General view of SURBOUM installation.

FIG. 5. SURBOUM: inside view.

5.4. Active cleaning enclosure

Sodium wastes are treated with water atomization in an inert gas flow. The inert gas acts as a coolant and prevents hydrogen ignition by diluting it. The waste is placed in special perforated metal baskets. This preparation is a prime necessity to obtain an efficient treatment. All sodium must be easily accessible to water.

This process allows a control of the effluents. Depending on the continuous hydrogen concentration measurement in the effluents, operator can drive the water flow. This process can be automated.

The active cleaning enclosure used at the CEA has a diameter of 3 meters and can accept 1000 kg of metallic wastes per load. This process is efficient for sodium waste treatment.

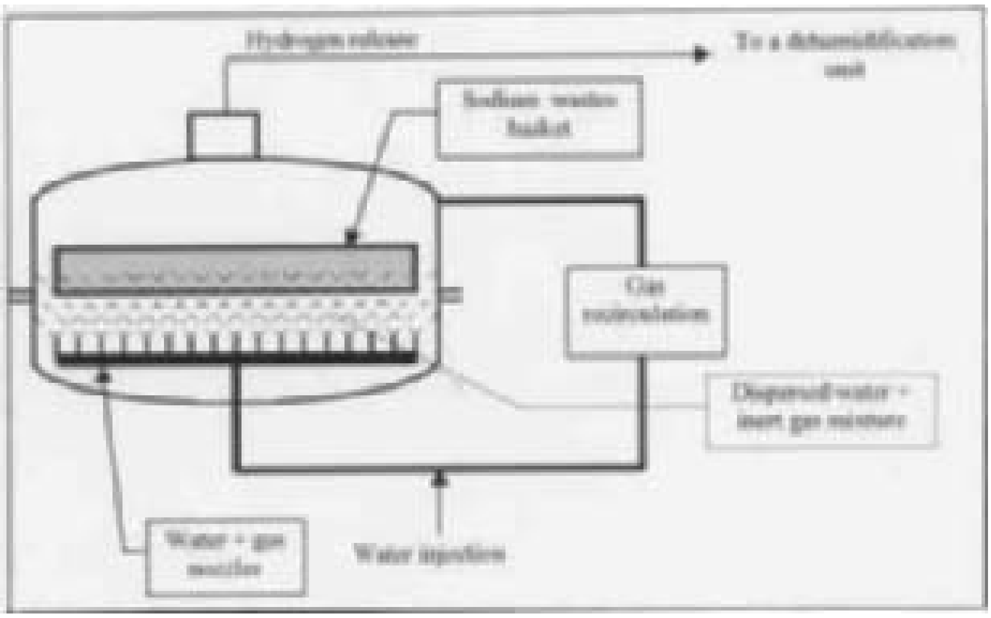


FIG. 6. Active cleaning enclosure process.

5.5. Results of the comparison

The autoclave process is not approved for a sodium loop decommissioning system, given the limited volume of sodium admissible. The comparison made here does not allow us clearly to point out and specifically approve one type of process between the active cleaning enclosure and the water jet treatment, that could be adaptable to all configurations on the work site. The use of cleaning pits seems to offer limited possibilities as a unique installation given the relatively limited dimensions of facilities of this type (at least for present day LMFR facilities), and also the admissible sodium quantity for each processing phase. This process can be used as a complementary installation for the treatment operations. The water jet treatment process appears to suit the processing of inactive sodium best. Moreover, this process presents the advantage of being relatively economical and fast. However, its use is more limited when dealing with active sodium. In such a case, the active cleaning enclosure process would be the most effective. Although the treatment phase is relatively slow, effluents are more easily controllable than with the water jet process.

6. FACILITY DISMANTLING

No matter which process is approved, the treatment stage is the limiting phase in the majority of systems. Since cutting precedes this stage, it must be limited to treatable quantities so that chain line treatment "clogging" can be avoided. In order to increase the rate of dismantling, a treatment stage allowing optimal performance must be found (For example, the construction of a double capacity treatment facility). The diagrams on the next page show as examples two "ideal" dismantling facilities for a large system, with the water jet process and the active cleaning enclosure process. Such facilities would be able to process respectively 10 m^3 and 2 m^3 of sodium waste per day. These two diagrams are proposed for a decommissioning operation equivalent to the treatment of secondary loops of reactor plant (Phénix or Superphénix size).

7. CONLUSION

This survey has enabled us to assess how the CEA has dealt with its inactive sodium loop decommissioning. Furthermore, it has permitted an extrapolation to active systems. The comparative assessment of cutting techniques and processes has put forward a primary selection of the best technologies to be used in future LMFR decommissioning.

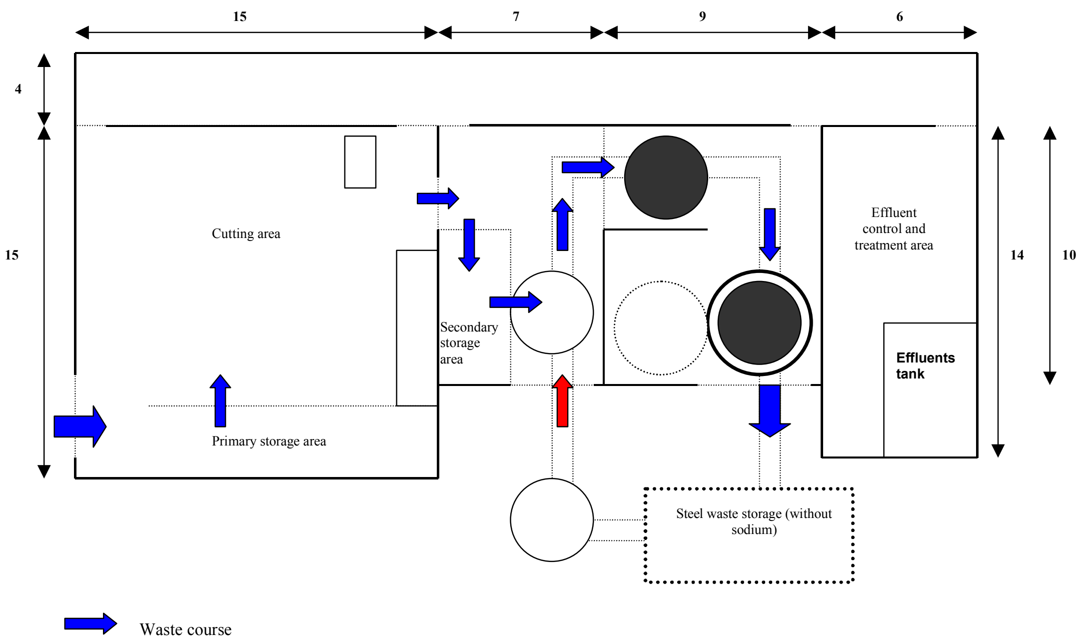


FIG. 7. 2 m^3 per day dismantling facility (active cleaning enclosure process).

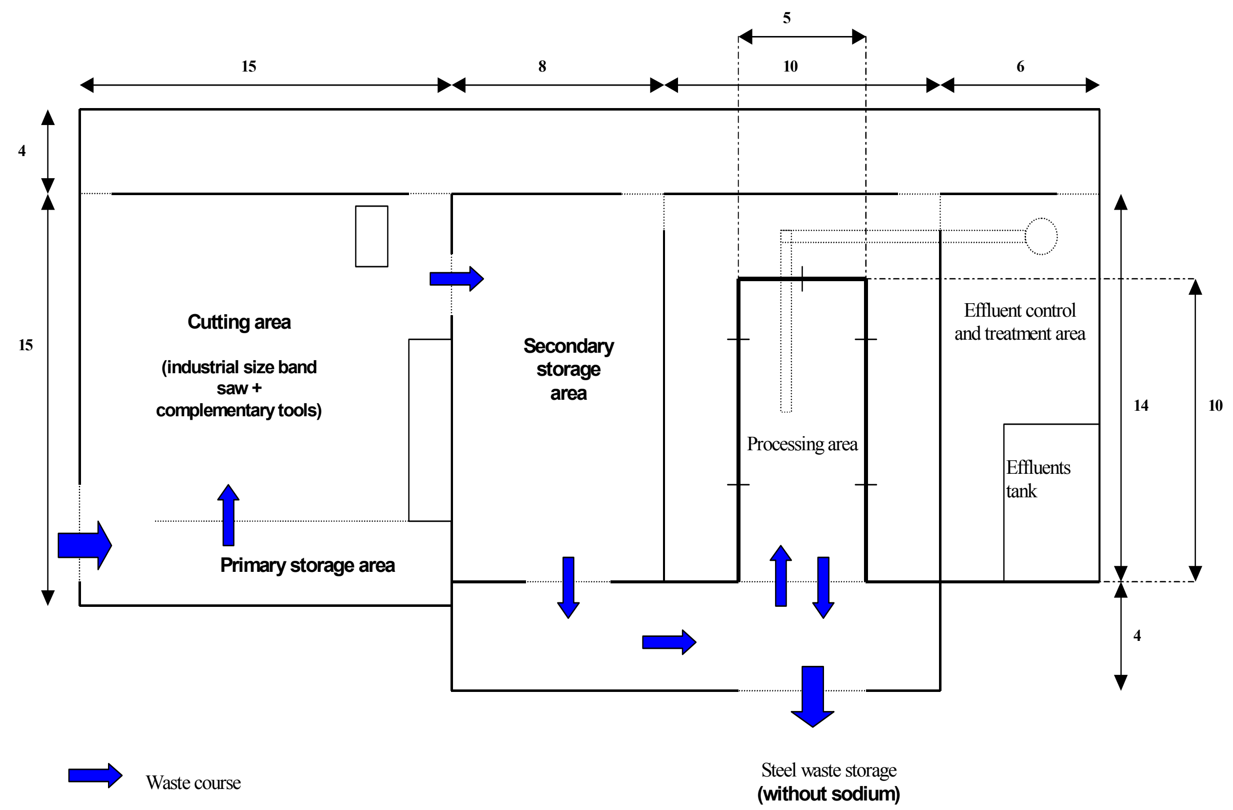


FIG. 8. 10 m^3 per day dismantling facility (water jet treatment).

REFERENCES

[1] ROGER, J., Stage 2: Dismantling of reactor. Case of experimental FBR RAPSODIE, paper presented in the International Symposium on Decontamination and Decommissioning, Knoxville, USA, 1994.

[2] COSTES, J.R., ANTOINE, P., GAUCHON, J.P., Decontamination before dismantling a fast breeder reactor cooling system, paper presented in the IAEA Specialists' Meeting on Sodium Removal and Disposal from LMFRs in Normal Operations and in the Framework of Decommissioning, Aix en Provence, France, 1997.

[3] RODRIGUEZ, G., GOEDERT, N., COURNILLE, M., Development of cutting techniques applied on sodium wastes. Application to Liquid Metal Fast Reactors decommissioning, Proc. 8th Intl. Conf. on Radioactive Waste Management and Environmental Remediation (ICEM'01),Brugess, Belgium, 30 September – 4 October 2001, Anibal Taboas, Rik Vanbrabant, Gary Benda (Eds), New York, ASME, TD898.14.M35 I57 2001 (2002).

[4] PILOT, G., BERNARD, J., COSTE, J.R., GRANDJEAN, J.P., Evaluation of steel cutting techniques, paper presented in the International Conference on Decommissioning of Nuclear Installations, Luxembourg, 1994.

[5] RODRIGUEZ, G., MASSE, F., ROBIN, J.C., BERTE, M, BUCHWALDER, M., SERPANTIE, J.P., Sodium waste treatment: from process development to industrial application, Proc. Intl. Conf. On Safe Management and Disposal of Nuclear Waste (SAFEWASTE 2000), Montpellier, France, 1-5 October 2000, Sociéte Française d' énergie nucléaire (SFEN), Paris, (2000) pp. 346-356.

[6] RODRIGUEZ, G., SAROUL, J., ARNAUD DES LIONS, J.P., Methods for sodium waste treatment coming from liquid metal fast reactors, Proc. World Nuclear Congress and Expo (ENC'98) Trans., Nice, France, 25-28 October 1998, Vol. IV, Workshops, (1998) pp. 85-89.

[7] MARMONIER, P., DEL NEGRO, R., Information about the accident occurred near RAPSODIE (1994, March 31^{st}), paper presented in the IAEA Specialists' Meeting on Sodium Removal and Disposal from LMFRs in Normal Operations and in the Framework of Decommissioning, Aix-en-Provence, France, 1997.

[8] MINGES, J., CHERDON, W., SCHUTZ, W., Short report of an accident during sodium cleanup with ethyl carbitol in a storage tank of a research facility, paper presented in the IAEA Specialists' Meeting on Sodium Removal and Disposal from LMFRs in Normal Operations and in the Framework of Decommissioning, Aix en Provence, France, 1997.

FAST REACTOR PHYSICS AND ENGINEERING EXPERIMENTS AND ANALYSES

(Session 3)

ANALYSIS OF SUPERPHÉNIX AND PHÉNIX NEUTRON PHYSICS EXPERIMENTS WITH THE ERANOS CODE AND DATA SYSTEM

J. TOMMASI, S. CZERNECKI, G. RIMPAULT, F. VARAINE
Commissariat a l'Energie Atomique (CEA), Saint-Paul-Lez Durance Cedex, France

Abstract

The neutron physics commissioning tests for Super-Phénix and some tests in Phénix have been re-evaluated using the recent ERANOS-1.2 code system and the ERALIB1 adjusted nuclear data library based on the JEF-2.2 evaluated data file. Compared to the older code and data system, agreement with experiment excellent and is obtained without the need to apply numerous corrections due to method biases and data poor accuracy.

1. INTRODUCTION

The start-up and operation of the Na-cooled fast reactors Phénix and Superphénix has provided a large amount of valuable experience, as the decommissioning of Superphénix is doing right now. We shall focus here on the feedback of the neutron physics experiments performed in both reactors on the neutron physics calculation tools and methods.

An extensive review of the Superphénix commissioning tests has already been performed [1] and deals not only with neutron physics tests but also with thermal-hydraulics, neutron monitoring, dynamic behaviour, failed fuel detection and decay heat tests.

We present here recent calculation results obtained with a new version of the neutron physics code system, the ERANOS-1.2 package [2]. Among the tools available are a versatile cell calculation module, ECCO [3], able to deal with complex geometries and to accurately solve the slowing-down equations in a fine-group scheme (collision probability method in many groups using the sub-group method), core flux solvers in diffusion or transport theory, 2D and 3D geometries [4 – 6], perturbation and sensitivity analysis tools, used in particular to create equivalent homogeneous cross-section sets for control rods. Nuclear data are taken from the JEF-2.2 evaluated data file [7].

A statistical adjustment procedure on the 17 most important nuclides (uranium, plutonium, steel components, oxygen, sodium), involving some 350 integral data measured in reactor and mock-up facilities, has lead to the ERALIB1 adjusted library [8], with reduced biases and uncertainties.

2. CRITICAL MASS

In Superphénix, the criticality of the first critical core (C1D for "cœur de première divergence") has been made with the control rods withdrawn from the core. Later on, another core (CMP for "cœur de montée en puissance") has been established in order to achieve full power.

Different core situations have been measured, with the control rods in different positions. There is a control rod system (CR), made of two rings: an inner ring (IR) made of 6 control rods, and an outer ring (OR) made of 15 control rods. And there are three shutdown rods (SR). The layout of the core can be seen in Fig. 1.

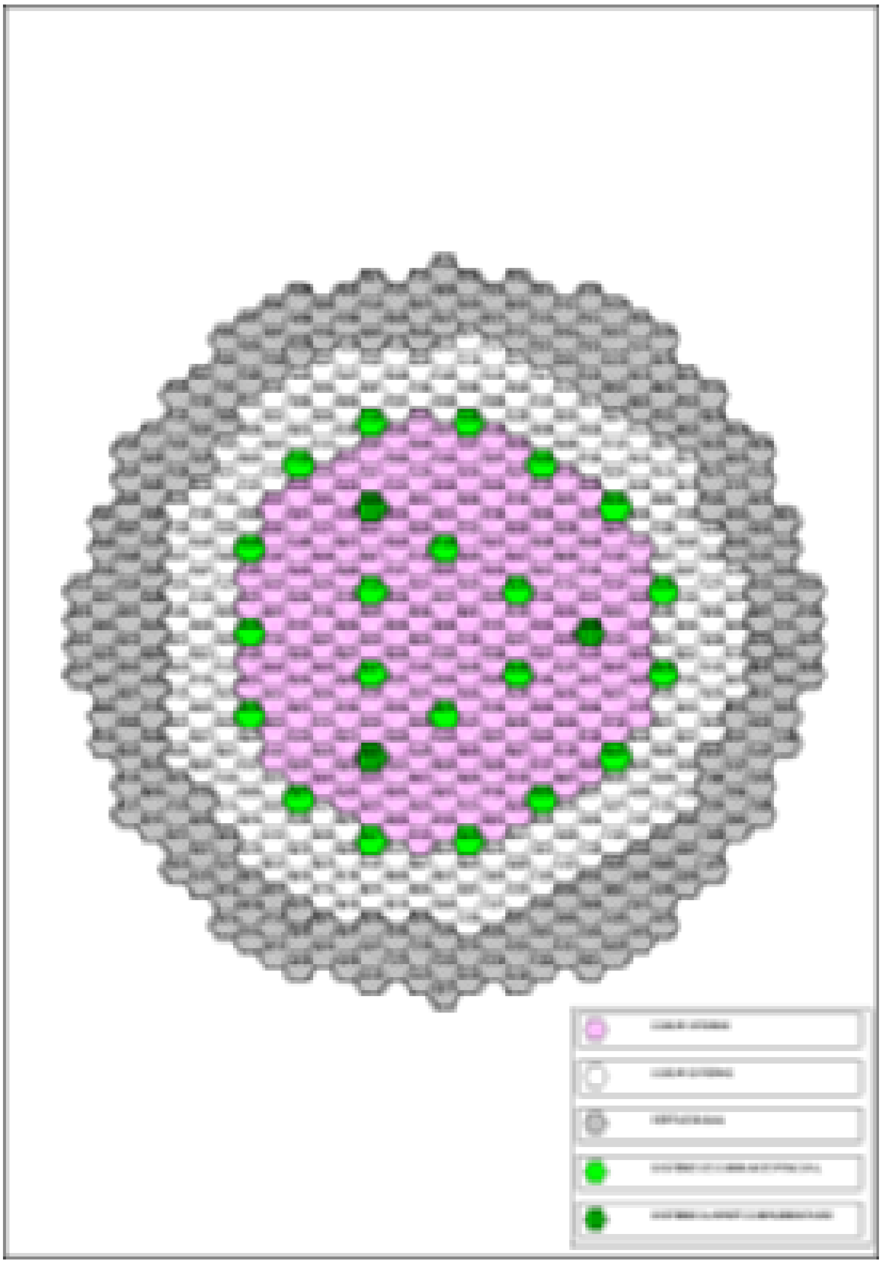


FIG. 1. Superphénix core layout.

Critical masses, CMP, cold shutdown at 180°C are shown in Table 1.

TABLE 1. CRITICAL MASSES, CMP, COLD SHUTDOWN (180°C), 1985

Inner ring Outer ring Shutdown rods	Critical height Critical height Extracted	Fully inserted Critical height Extracted	Critical height Fully inserted Extracted	Extracted Extracted Extracted
Experiment E (pcm)	0	0	0	3710
Calculation C (pcm)	70	56	24	3762
E-C	-70	-56	-24	-52

The results given depend on the reactivity worth of the control rods. It will be shown in the next section that the prediction of the control rod worth, using the reactivity equivalence method, is also very satisfactory. The calculations have been performed using an accurate scheme in the ECCO cell calculations, with a fully heterogeneous 2D cross-cut of the assembly allowing to take into account the wrapper heterogeneity effect (approx. 250 pcm) and an accurate fine-group treatment for the slowing-down (1968 energy groups) before condensation to the 33 energy groups core calculation scheme. Core calculations have been performed with the 3D variational nodal transport solver VARIANT in the ERANOS code system. The nuclear data come from the ERALIB1 adjusted file. It is worth noting that the results are obtained with no correction applied, while with the older calculation scheme large method biases (approx. 1400 pcm, due to transport effects, mesh effects, heterogeneity effects, nuclear data) had to be applied to the raw calculation results in order to recover a good agreement with experimental values.

3. CONTROL ROD WORTH

An extensive experimental programme has been carried out with the aim of precisely defining the values of the different control rods and their interaction effects. The measurements have been performed with the reactor in a sub-critical state by using counters placed at the centre of the reactor and at 3 different locations under the reactor vessel. The responses of these last 3 counters were amplified by neutron guides. This type of sub-critical measurement requires correction factors to account for the fact that the counters see a different perturbation depending on their position in the reactor. These correction factors have been calculated using different methods and data and show no dispersion in their results and so these corrections have therefore not been recalculated.

The preparation of cross sections for control rod absorbers requires a special treatment due to the very high coupling of the heterogeneous control rod structure to the surrounding core cells. The method used is the reactivity equivalence method [9] which has been validated on the BALZAC 1H experiments performed in the MASURCA zero-power critical mock-up in Cadarache [9]. This method uses the Sn transport option of the BISTRO code and its associated perturbation modules. The validity of such an approach for control rods has been evaluated not only for the reactivity variation of the control rod but also for the absorption rates in the control rod and the surrounding core regions.

Table 2 shows the control rod homogenisation techniques (SPX control rods model), and Table 3 shows the control rod homogenisation techniques 9SPX lower backup rods model, respectively).

TABLE 2. CONTROL ROD HOMOGENISATION TECHNIQUES (SPX CONTROL RODS MODEL)

	Heterogeneous (reference)	Homogeneisation by volume, %	Homogenisation by flux, %	Homogenisation by equivalence, %
Antireactivity (pcm)	8846	+24	+6.8	-0.1
Absorption rate	0.345	+24	+7.0	-2.0

TABLE 3. CONTROL ROD HOMOGENISATION TECHNIQUES (SPX LOWER BACKUP RODS MODEL)

	Heterogeneous (reference)	Homogeneisation by volume, %	Homogenisation by flux, %	Homogenisation by equivalence, %
Antireactivity (pcm)	6401	+23	+4.2	0.0
Absorption rate	0.258	+23	+4.2	-1.0

One can see the validity of the recommended method (errors in the last column) but also the inadequacy of simpler methods (homogenisation by volume or by flux) for the determination of the control rod worth. The control and shutdown systems have been measured under different situations and their control rod reactivity worth have been calculated with the current scheme. The results are given in Table 4.

TABLE 4. SUPERPHÉNIX CONTROL ROD WORTH (CMP, 180°C, 1985)

	Exp. (pcm)	Cal. (pcm)	C/E
$\Delta\rho$(CR), SR extracted	8067 ± 995	8119	1.006
$\Delta\rho$(SR), CR fully inserted	1193 ± 155	1115	0.935
$\Delta\rho$(SR), CR at critical height	1039 ± 120	1009	0.971
$\Delta\rho$(3 outer rods) other CR at critical height	-1115 ± 134	-1106	0.992
$\Delta\rho$(3 inner rods) other CR at critical height	-1530 ± 184	-1505	0.984

Results with the previous calculation scheme were giving C/E values of about +5 to 20%. The fact that the comparison is significantly improved indicates that both the method and the nuclear data have been improved and that there remain probably no compensating effects. The results obtained with the ERANOS calculation scheme are therefore satisfactory and this scheme can be considered to be reliable as an explicit treatment of all shadowing and transport effects is taken into account. This good behaviour is also observed for the prediction of control rod reactivity worth in the Phénix reactor. There is only a ring of 6 control rods in this reactor. The results are gathered in Table 5.

TABLE 5. PHÉNIX CONTROL ROD WORTH
(REACTIVITY measurements, 1995) with β_{eff} = 339 pcm

	Exp. (pcm)	Cal. (pcm)	C/E
$\Delta\rho$(all 6 CR)	7892 ± 359	7752	0.98
$\Delta\rho$(CR n°1)	1200 ± 61	1205	1.00
$\Delta\rho$(CR n°4)	1180 ± 61	1190	1.01
$\Delta\rho$(CR n°6)	1268 ± 64	1251	0.99

4. POWER MAP DISTRIBUTION

The power map distribution is deduced from the measurement of the sodium temperature at the output of each reactor sub assembly. This is not a purely neutron physics experiment and assumptions on the way the sodium flows in the sub assembly are of major importance in this measurement. However after checking the various sources of uncertainties, only a correction for the mixing of sodium from different sub assemblies before detection by the thermocouple has been found to be significant.

Discrepancies with the previous calculation scheme were as large as 17% if the differences between the values at the centre of the reactor (averaged over the first 5 fuel S/A rings) and the values at the edge of the reactor were considered. It was recognised that some discrepancies could arise from nuclear data as was shown by sensitivity calculations as well as inter comparisons between different cross-section sets. However it is only recently that the reasons of such a discrepancy have been found. The treatment of both mesh and transport effects was leading also to a significant misprediction of the power map distribution. Finally the determination of control rod heterogeneity effects was making a significant contribution to the observed discrepancies.

With the new ERANOS calculation scheme and data, all of these effects are treated explicitly and, provided that the cross sections are sufficiently accurate and that there are no compensating errors (the power map distribution and the reaction rate distributions of a large core have not been used in the adjustment), the comparison should be satisfactory.

Figure 2 shows the discrepancies averaged over each ring of fissile subassemblies, either with the old calculation scheme and data (A) or the new ones (N).

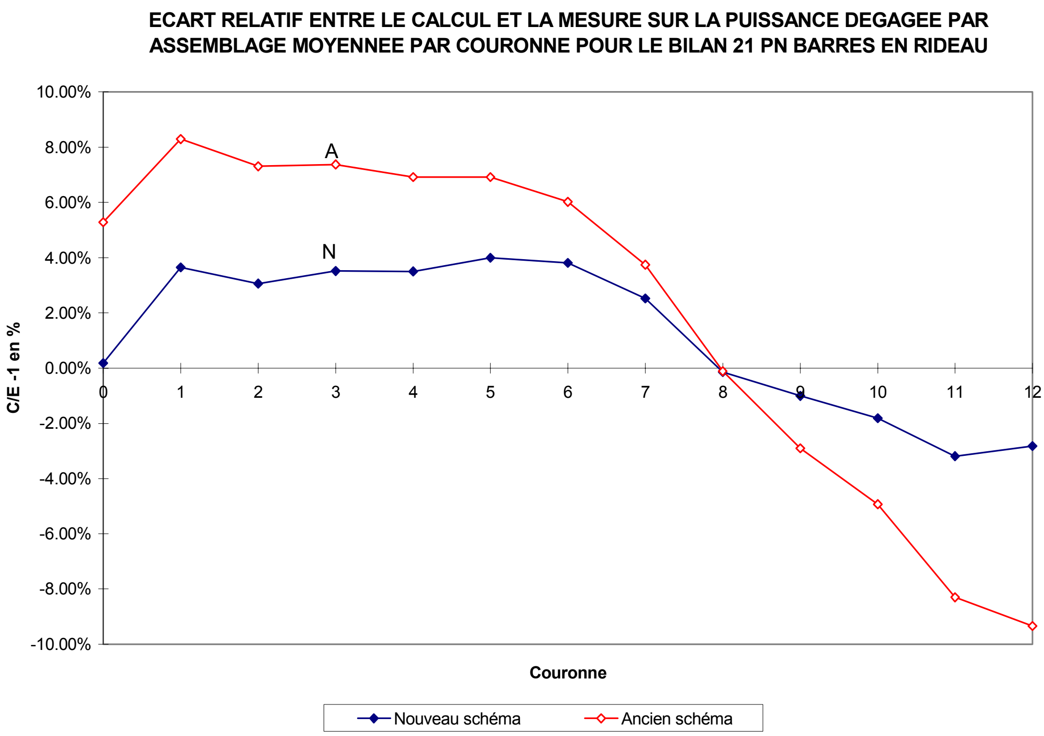


FIG. 2. Discrepancies on S/A power output, averaged over each ring of S/As old (A) and new (N) calculation scheme + data.

Figure 3 shows the spread within each ring of the discrepancy between calculation and experiment. The main source for the discrepancy reduction between the old and new calculation scheme lies in nuclear data (6%), the nuclides involved being ^{239}Pu, ^{238}U, ^{56}Fe, and oxygen. Method biases (mesh and transport effect, radial temperature gradient for the fuel) account for other 5% in the global discrepancy gradient (defined as $G = (C/E)_{5\ inner\ rings} / (C/E)_{last\ fuel\ ring} = 1.18$ with the old scheme, 1.05 with the new one). A correction on the measured temperatures to take into account sodium recirculation leads to a further 3% correction on G. Nevertheless, a small residual bias remains, as well as large spreads over each S/A ring.

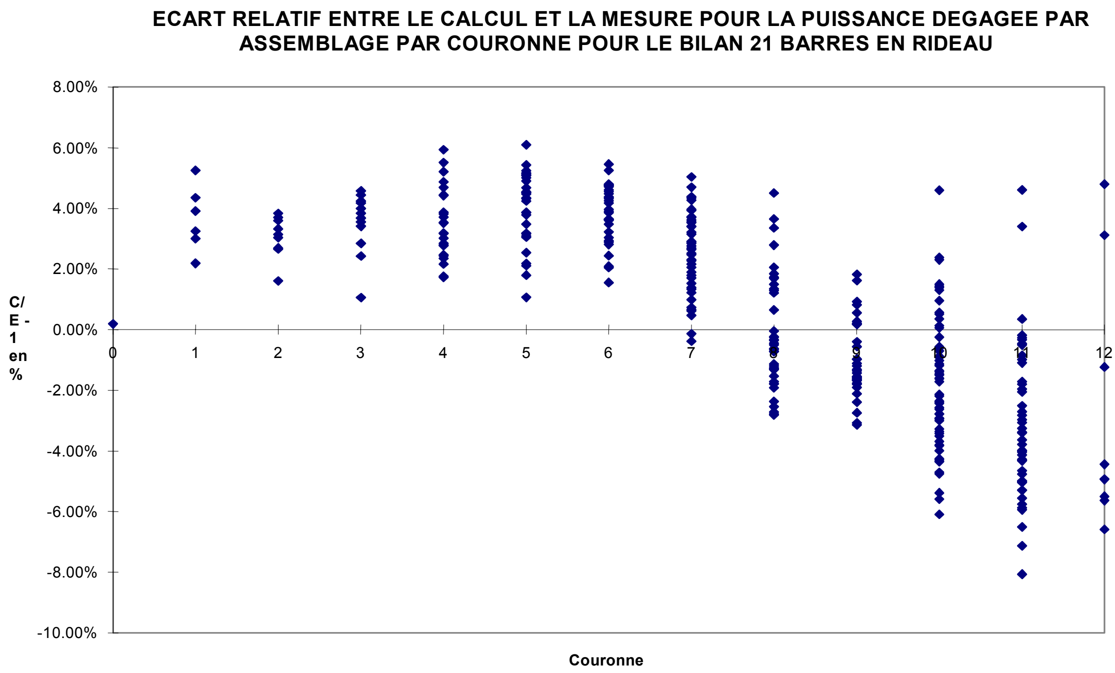


FIG. 3. Spread of the discrepancies on S/A power output over each ring of S/As new calculation scheme + data.

Another experiment related to the power map distribution was the irradiation of foils in a row of subassemblies placed on a core radius (these are the T1 and T2 experiments: T1 with all control rods at the same height, T2 with the outer ring less inserted than the inner one). The foils allowed the measurement of ^{239}Pu, ^{238}U, and ^{235}U fission rates, and of ^{238}U capture rates. From these, the radial power distribution could be inferred. The results are shown in Figs 4 and 5.

It can be seen that the power map is correctly reproduced as the discrepancies, remaining less than 2%, remain within the experimental uncertainties. An example of axial distributions is given in Fig. 6, showing as well a good agreement between calculation and experiment.

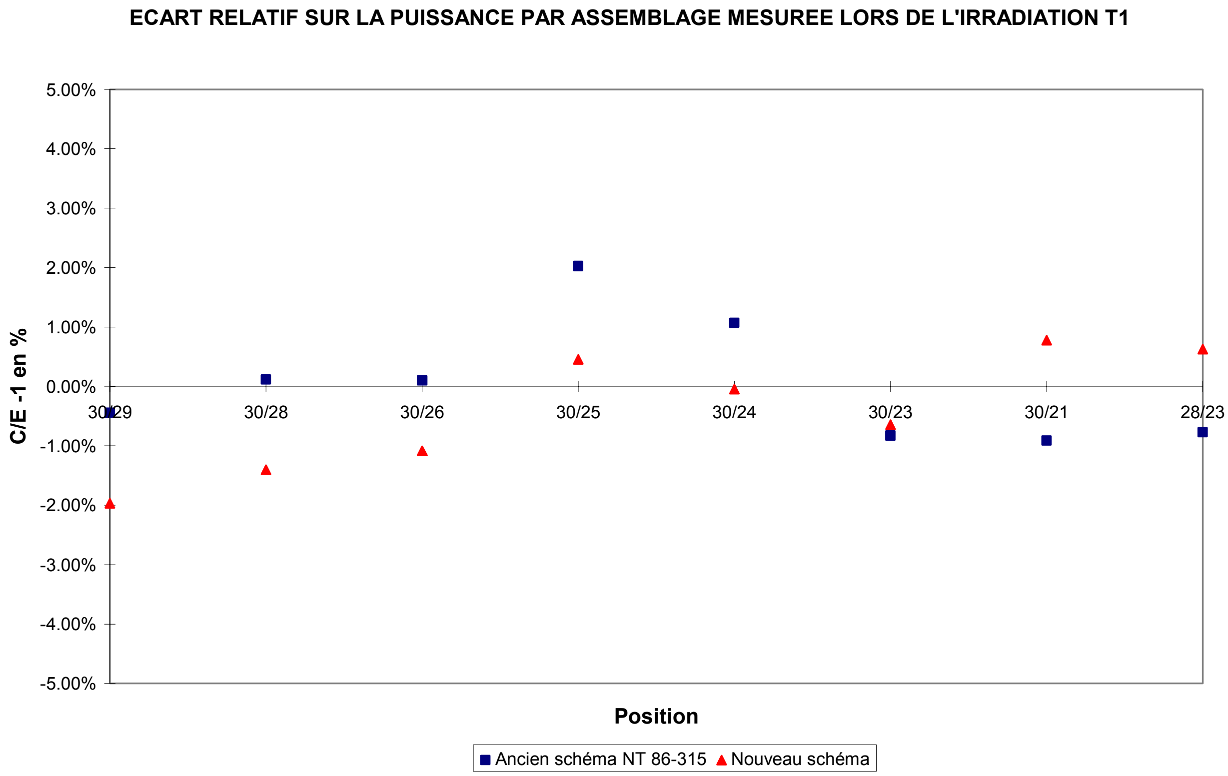


FIG. 4. T1 foil irradiation: calculation and experiment, old and new calculation schemes.

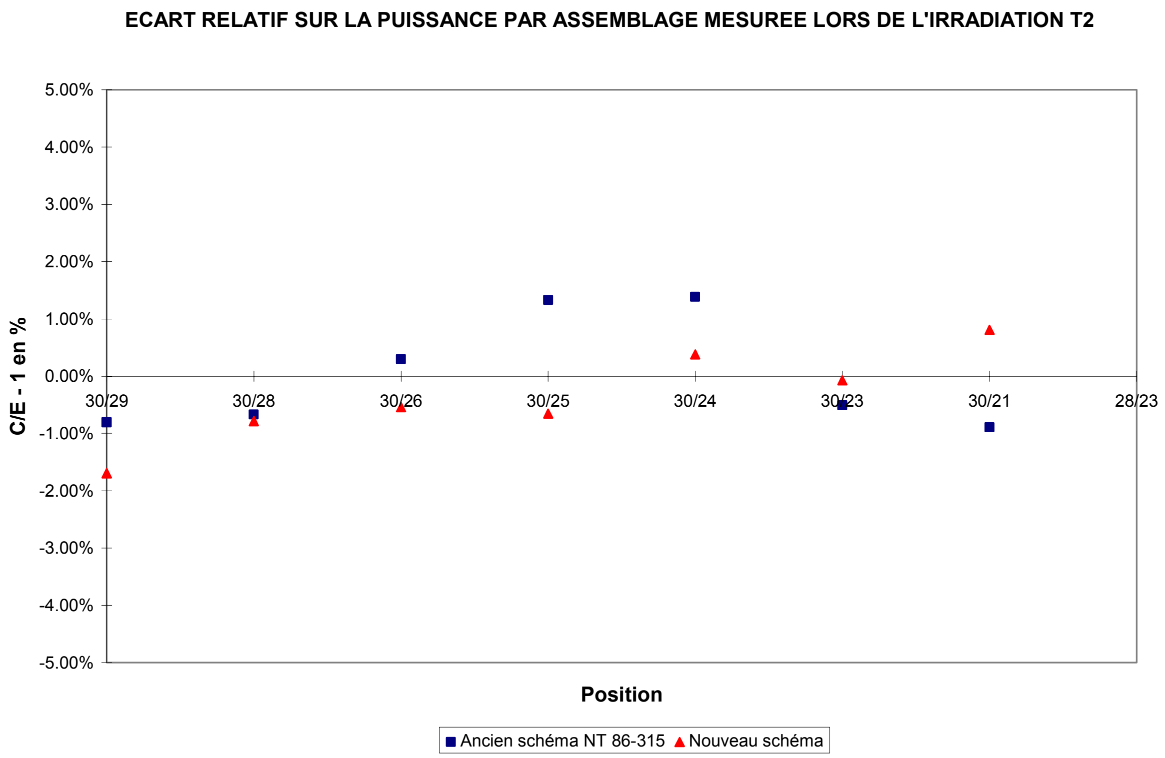


FIG. 5. T2 foil irradiation: calculation and experiment, old and new calculation schemes.

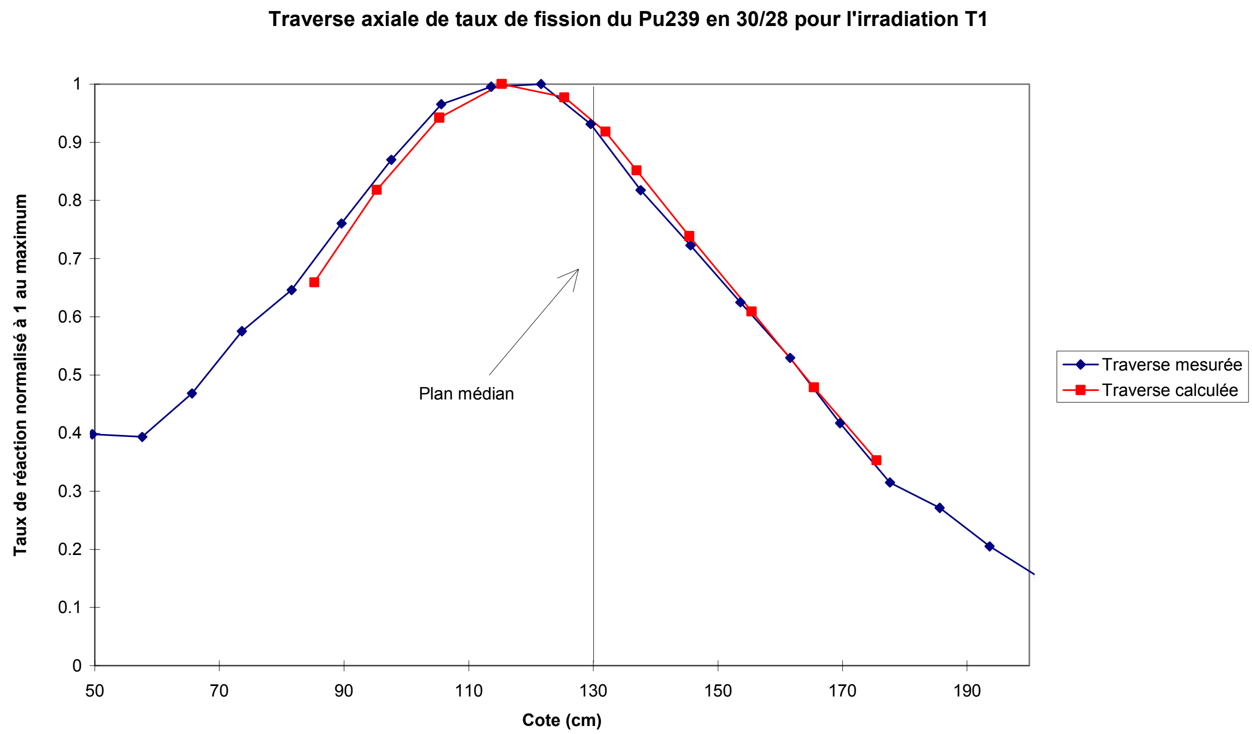


FIG. 6. T1 foil irradiation: axial distribution, Pu239 fission, S/A 30/28.

5. BURNUP REACTIVITY SWING

The burnup reactivity swing has been deduced from the reactivity balances at different moments during the operation of the reactor. Control rod insertions as well as temperature increases have been taken into account to deduce the measured values (using the S curves of the control rod system). Advantage has been taken from the fact that the reactor has had periods without operation to separate the reactivity variation with time into two components: the loss of reactivity caused by burnup (heavy nuclide transmutation and fission due to the running of the reactor under power), and the loss of reactivity due to the natural ^{241}Pu decay. The experimental results used for the comparison with calculation are taken from the first period of the life of the reactor (82.3 FPD over the first 40 months).

The previous calculation scheme gives C/E values of 0.85 for the burnup component and 0.94 for ^{241}Pu decay. The burnup reactivity swing is therefore significantly under-predicted. Considerable attention has been given to the different components entering into the calculation, which include the fission product cross-sections and yields, the heavy nuclide component (fission and capture cross-sections), the energy release and the migration of gaseous fission products into the plenum.

The same analysis has been performed with the new calculation scheme and data. The fission products can be taken into account explicitly, with 87 individual fission products, or by the means of 6 lumped fission products. The comparison to experiment is as follows:

TABLE 6. BURNUP REACTIVITY SWING, C/E VALUES

	18 heavy nuclides 87 fission products 5 day for migration of gaseous FPs	18 heavy nuclides 6 lumped fission products no gaseous fission product
Burnup component	0.94	0.93
^{241}Pu decay component	0.80	0.80

The calculation results are satisfactory even when using simplified calculation schemes with lumped fission products. The reasons for the improvement in the burnup component are linked to differences in the energy release per fission and the fission cross section of Plutonium-239, which for the same power normalisation induce a different fluence. Also of importance are the fission product cross sections and the fact that complete decay chains are explicitly treated. However the decay component remains mispredicted.

6. DOPPLER REACTIVITY COEFFICIENT

During the start-up of Superphénix, an experiment related to the Doppler effect has been performed, on the CMP core, decreasing slowly the temperature from 400 to 180°C while maintaining isothermal conditions in the reactor. The increase in reactivity was compensated by control rod insertion. The contributions of the expansion reactivity coefficient (linear with respect to temperature) and of the Doppler effect (logarithmic with respect to temperature) have been separated. The model took into account the effective temperature, using the Debye temperature. The comparison of experiment and calculation, using the reference scheme is given in Table 7.

TABLE 7. DOPPLER CONSTANT MEASUREMENT AND CALCULATION

Measured Doppler constant	1240 ± 175
Calculated Doppler constant	1243
C/E	1.00

7. CONCLUSION

This paper presents the results obtained with the ERANOS calculation scheme and data for several major parameters of the Super-Phénix start up core (CMP) including the critical mass, the control rod reactivity worth, the power map distribution, the burn up reactivity swing and the Doppler feedback. Agreement of calculation results with experiment are very satisfactory: agreement within 100 pcm for the critical mass, less than 5% discrepancy on control rod worth, a residual radial gradient of less than 5% the radial power map, an discrepancy less than 10% on the reactivity swing, and a full agreement on the Doppler constant.

This demonstrates a significant improvement on the predictions made with the previous calculation scheme and data, in particular by calculating the values of the core experiments directly without the need for separate calculation and application of numerous additional corrections (e.g. the control rod worth calculation with the old scheme and data required some 50 basic calculations to reach the method biases necessary, while only 3 are needed with the new tools).

REFERENCES

[1] Nuclear Science and Engineering, **106** (special issue devoted to SuperPhénix commissioning tests) (1990) pp. 1–98.

[2] DORIATH, J.Y., Mc CALLIEN, C., KIEFHABER, E., WEHMANN, U., RIEUNIER, J.M., ERANOS 1: the advanced European system of codes for reactor physics calculation, Proc. Intl Conf on Mathematical Methods and Super Computing in Nuclear Applications, Kernforschungszentrum Karlsruhe, Germany, 19–23 April 1993, D.H. Kusters, E. Stein (Eds), Kernforschungszentrum Karlsruhe (1993).

[3] RIMPAULT, G., Algorithmic features of the ECCO cell code for treating heterogeneous fast reactor assemblies, Proc. Intl Conf on Mathematics and Computations, Reactor Physics, and Environmental Analyses, Portland, Oregon, USA, 1–5 May 1995, American Nuclear Society (ANS), ISBN 0-89448-198-3 (1995).

[4] PALMIOTTI, The BISTRO optimised two-dimensional Sn transport code, Nuclear Science and Engineering 104 (1990) pp. 26–33.

[5] CARRICO, C.B., LEWIS, E.E., PALMIOTTI, G., Three dimensional variational nodal transport methods for Cartesian, triangular and hexagonal criticality calculations, Nuclear Science and Engineering, **111** (1992) pp. 168–179.

[6] PALMIOTTI, G., CARRICO, C.B., LEWIS, E.E., Variational nodal methods with anisotropic scattering, Nuclear Science and Engineering, **115** (1993) pp. 233–243.

[7] ROWLANDS, J.L., TUBBS, N., The EF-2.2 Nuclear Data Library, JEF-DOC 17, Nuclear Energy Agency of the OECD, April 2000.

[8] FORT, E., ASSAL, W., RIMPAULT, G., ROWLANDS, J., SMITH, P., SOULE, R., Realisation and Performance of the Adjusted Nuclear Data Library ERALIB 1 for Calculation Fast Reactor Neutronics, paper presented in the Conference on Physics Reactor Neutronics (PHYSOR'96), 16–20 September 1996, Mito, Japan.

[9] CARTA, M., GRANGET, G., PALMIOTTI, G., SALVATORES, M., SOULE, R., Control Rod Heterogeneity Effects in Liquid Metal Fast Breeder Reactors: Method Developments and Experimental Validation, Nuclear Science and Engineering, **100** (1988) pp. 269–276.

PRESERVATION OF FAST REACTOR KNOWLEDGE AND EXPERIENCE

(Session 4)

R&D LMFRs KNOWLEDGE PRESERVATION FRENCH PROJECT

F. BAQUE
Commissariat a l'Energie Atomique (CEA), Saint-Paul-Lez Durance Cedex, France

Abstract

The French Institutions involved in LMFRs development (EDF Utilities, CEA Research Institute and FRAMATOME-ANP Engineering) decided in 2000 year to preserve the R&D knowledge which was raised during the last four decades of the 20th century: the long term availability (some decades) of LMFRs experience should be maintained thanks to an extensive, everlasting and intelligible form which could allow future designers to use this great amount of knowledge. Among different types of architecture, the LMFRs Fund of Knowledge is being raised in 2001 and 2002 along two complementary ways:

- The main one gives an overall vision of R&D work undertaken through 23 R&D items: an overall view of LMFRs conception; 9 items about Core R&D work (Safety, Working, Thermohydraulics, Thermomechanics, Design Rules, Materials, Fuel clad failure, Neutronics, Nuclear fuel; 13 items about Reactor R&D work (Safety, Working, Sodium Coolant, Sodium Technology, Thermohydraulics, cover gas Aerolics, Thermomechanics, Design Rules, in sodium equipment Mechanics, Materials, In Service Inspection and Repair, Sodium Fires, Decommissioning).
- The other one deals with the Design of what can be a LMFR: Superphénix Plant was chosen as the largest and validated industrial size LMFR through the conception of its 41 systems (Core system, Monitoring systems, Protective and Shut-down systems, Primary systems, Secondary and Steam Generator Systems, Decay Heat Removal system, Primary and secondary Handling systems, Cleaning and Decontamination systems, Fuel Storage system).

Each R&D item and each Superphénix system is described with a Documentary Form, written by French specialist: after a brief description of the different sub-items (some pages), the list of relevant references are listed (some dozens to some hundreds of synthesis reports, basic literature, specialist interviews, Superphénix measurement data, EFR Project synthesis…). Thus, the LMFRs Fund of Knowledge is made of the 64 Documentary Forms and of all the relevant bibliography (more than 10 000 documents). It is made of paper documents and of computerised CD-ROMs. Bibliographic research can be made easily thanks to computerised organisation. The Fund is recorded under safe conditions by the CEA Record Office. Of course, accessibility and up dating are strictly limited.

1. INTRODUCTION

After four decades of R&D work, and also of design, construction, operation and decommissioning of prototype LMFRs (Rapsodie, Phénix and Superphénix1 reactors, Superphénix2, 1500Project and EFR projects), the absence of French and European future project highlighted the risk of loss of knowledge in various fields (excepted maybe decommissioning aspects). As R&D work came in the same time to its end, a number of synthesis reports were written and it became obvious that an overall preservation work had to be undertaken.

Thus, the French nuclear partners (CEA Research Institute, EDF electrical Utilities, FRAMATOME-ANP Engineering Compagny) have decided in 1999 to perpetuate this huge amount of knowledge and experience which have been acquired on LMFRs.

The aim is to create an everlasting data base of documentation, with easy access for consulting, in order to help the LMFR designers of the next future (in 2040…): help for understanding the design and technical options which were selected in the 20th century (coolant, pool or loop type, fuel…), so that they will be able to make properly their own choice and face their R&D need.

In that way, only consistent documents have to be selected by specialists, to not overcrowd such a base with worthless information.

2. AVAILABLE DATA

The French sodium LMFRs knowledge is based on a large and consistent background: four decades of R&D work (1960-2000), on all the fields of sodium nuclear plants: neutronics, safety, thermalhydraulics, mechanics, sodium technology… A huge database lies in test results, modelling effort and licensing documents. A number of specific databases exists in different services of the CEA and can be extensively used for the preservation project.

For one decade, the ACCORE system has been specified by the CEA in order to preserve and provide easy access to the experience gathered during nuclear studies (principally for LMFRs but also for PWRs): the ACCORE team deals with the management of technical documentation, identify experience underlying to existent documents, find out non-written experience through specialist interviews and, last but not least, provide an interactive access to knowledge for every CEA agent. Of course, this system is of a great help for the Documentary Fund project.

The industrial feedback of existing LMFRs is also of great interest because it gives a high level of qualification to R&D work and to design options, thanks to operating conditions. Rapsodie first prototype allowed feasibility tests from 1967 to 1981, Phénix NPP is still running until 1974 and Superphénix1 NPP was the largest worldwide LMFR between 1985 and 1998.

After two intermediate steps (Superphénix2 and 1500 Project), the EFR project allowed to finalize in the 90s a full detailed European LMFR design.

Due to the French cooperation between CEA, EDF and FRAMATOME-ANP LMFR specialists, the large range of R&D knowledge could be faced: a lot of specialists were still involved in the powerful Superphénix Plant and/or EFR Project in the end of the 90s.

In the same time, the international cooperation was strong enough to also enlarge the French specialist view and let them embrace all the variety of design and technical options (principally: DFR and PFR British Plants; BOR60, BN350 and BN600 Soviet Plants, Joyo and Monju Japanese Reactors).

3. BUILDING AN EVERLASTING DOCUMENTARY FUND

As soon it was decided to preserve the LMFRs knowledge, the specifications of the project were raised: type of architecture (see hereafter), type and number of selected documents, LMFRs specialist work, access and up to dating of the Fund, confidence of the information…

It was decided that the CEA specialists should work for R&D items dealing with the core and that the FRAMATOME-ANP specialists should work for R&D items dealing with the NPP and for Superphénix conception files.

According to a specific Quality Insurance Form, each specialist selected the consistent documents that he could find in his own literature fund and also through ACCORE system. He wrote a documentary sheet (see Section 5: a brief synthesis where sub-items and associated selected documents are listed. When it was considered that a strong synthesis was not available, the specialist wrote it.

About 20 CEA specialists, 10 FRAMATOME-ANP specialists and 6 EDF specialists were involved in writing and reviewing work.

After writing documentary sheets and, when necessary, synthesis documents, reviewing was perform by other specialists. The project team finally homogenized the documentary sheets and collected the documents.

The Fund will be made of about 15 000 paper documents (1 500 000 pages), of corresponding microfilms and files on CD-ROMs (both HTML and PDF type). When CD-ROM format will become unavailable, the data will be transferred on another support.

The Project team looked for all the selected documents then scanned them in order to obtain files and microfilms. Easy research will be possible through the Fund, using CD-ROMs, with exhaustive table of all the document characteristics (title, author, date of issue, place within the fund, key words). Active links exist within the data processing Fund: this modern approach avoids to deal with heavy catalogue of dispersed items (a former 'classical' LMFRs data base included about 250 items!).

The CEA Archive service will have the Fund in hand (papers, microfilms and CD-ROM) and keep it for the next decades in classical safe conditions, in order to assume the integrity of the Fund. EDF and FRAMATOME-ANP will have only the easy reproducing CD-ROM base.

Commercial to confidence is considered thanks to existing French LMFR Knowledge Procedure. Future addenda will be possible (mainly for dismantling item and for Phénix end of life phase).

The project took place from mid 2000 (specifications) to mid 2002 (fund available).

4. THE FUND ARCHITECTURE

Among different types of architecture, it was decided to simplify the Fund by sharing it in only two categories:

- R&D items that can be either scientific fields (neutronics, mechanics…) or polyvalent field (safety, design…);
- Conception files of Superphénix Plant (including R&D supporting work), considered as a large (industrial size) and qualified LMFR.

It is a simple way to preserve the great variety of knowledge (no huge subject catalogue) and, thanks to active links within the soft base (information processing), it will be possible to find any specific information which deals with any particular subject: for example, mechanical behaviour of core materials will lie in different R&D items (mechanics, safety) and Superphénix systems (core).

Of course, when looking after this information, the designer of the future will find a lot of documents and will have to organize, to select and to range them; but we think this is an important part of his job in order to an LMFR specialist of the future…

Among the selected documentation, some documents can be mentioned for their major interest: the Superphénix Plant Safety Report, the licensing documents (SYFRA system), the codes (only their documentation will be preserve: presentation, qualification, utilization, computer description), the RCC-MR rules (2000 version), LIMET’88 proceedings, ARCOPAC data base (Superphénix Plant measurement files) and EFR synthesis reports.

The overall architecture of the LMFRs Documentary Fund is finally:

- Overall view of the Fund;
- 21 R&D items:
  1) R&D items about the core:
     - Safety;
     - Thermalhydraulics;
     - Thermomechanics;
     - Codes and rules;
     - Materials;
     - Fuel clad failure;
     - Neutronics;
     - Fuel;

  2) 13 R&D items about NPP:
     - Safety;
     - Operational experience;
     - Sodium coolant;
     - Thermalhydraulics;
     - Cover gas aeraulics;
     - Thermomechanics;
     - Codes and rules;
     - Materials;
     - In sodium equipments;
     - In service inspection and repair;
     - Sodium fires;
     - Dismantling;

- Superphénix conception systems: 9 main systems (41 systems):
  1) Core;
  2) Plant monitoring;
  3) Primary loop;
  4) Secondary loop;
  5) Handling;
  6) Steam generators;
  7) Cleaning and decontamination;
  8) Fuel storage;
  9) Decay heat removal.

5. DOCUMENTARY SHEETS

The LMFRs Documentary Fund is based on the 63 above-mentioned items: each item is described within a Documentary Sheet.

Each Documentary Sheet includes:

- A brief presentation of the R&D item (or of the Superphénix system): objectives, processes, successive evolutions,…;
- A list of all the sub-items and connected items;
- The knowledge assessment for each sub-item;
- The possible future prospects (ideas for future R&D…);
- The bibliographical list (title, author, reference, date of issue, Fund index).

Through writing, reviewing ad project team final checking phases, one can consider that the state of the art could be reached. Of course, some specialists retired before this Fund was built, but their interviews, the existing documentation and so called 'lower rank' specialist work allowed to reach the goal.

6. CONCLUSION

After four decades of R&D, design and operation with LMFRs, and facing no project, CEA, EDF and FRAMATOME-ANP decided in 2000 year to preserve the LMFRs knowledge.

In the year 2002, this Fund is available: 63 items (overall conception, 21 R&D items and 41 Superphénix system conception files) have been considered. Consistent documentation has been selected (15 000 documents) and saved on paper, microfilm and CD-ROM medium.
Easy access, integrity and up dating are available.

Among the risks that could disturb the LMFRs Documentary Fund building, the main one was the specialist missing. Thanks to coordinated efforts of CEA, EDF and FRAMATOME-ANP teams, and to extensive use of ACCORE system data (documentation and interviews), one can consider that the French LMFRs knowledge is properly perpetuated.

R&D LMFRS KNOWLEDGE PRESERVATION FRENCH PROJECT: APPLICATION TO THE SODIUM COOLANT AND COVER GAS

G. RODRIGUEZ
Commissariat a l'Energie Atomique (CEA), Saint-Paul-Lez Durance Cedex, France

Abstract

In the frame of Liquid Metal Fast reactors knowledge preservation French project, CEA has separated the field of this subject in several items and has designed several experts to treat specific areas. Thus, this paper presents how this study was made dealing with the following item: sodium coolant and cover gas. This paper explains what were the main difficulties by making this study and how they can be solved.

1. INTRODUCTION

Development of R&D linked to Liquid Metal Fast Reactors had an important stop in France since the decision of the final shutdown of Superphénix in 1998. From the very beginning of this technology until now with the decommissioning aspects, France have accumulated a huge knowledge in this field mainly shared by three companies: CEA, EDF and FRAMATOME ANP.

In 2000, it has been decided to create an everlasting data base of documentation with easy access for consulting, in order to keep knowledge an experience acquired through the development, design, construction and operation of the French LMFR plants (RAPSODIE, Phénix, Superphénix, and EFR projects).

This database was considered to be important in order to perpetuate the knowledge on LMFRs for the following decades. The absence of French and European future project on this subject highlighted the risk of loss of knowledge in various field (excepted maybe decommissioning aspects), and expresses the need to drive as soon as possible such a project.

2. GENERAL PRESENTATION OF THE PROJECT

The R&D LMFRs preservation knowledge French project started in 2000. CEA is the leader project. The aim of this data base project is to include:

- Synthesis of history, concept approach, evolution and supporting R&D of French LMFRs;
- 22 R&D topics and 41 SUPERPHÉNIX plant systems;
- Description of the main common CEA, EDF and FRAMATOME ANP realizations;
- More than 1500 documents (papers and CD Rom medium);
- RCC MR 2000 version (updates of the mechanical analysis rules);
- Database of all the measurement recorded and obtained from SUPERPHÉNIX exploitation.

From the R&D part, the most important items are the following:

- Neutronic;
- Fuel;
- Fuel rupture;
- Sodium cooling and argon gas;
- Sodium technology;
- In sodium mechanics;

- Safety and operation;
- Thermohydraulic;
- In sodium inspection;
- Sodium fires;
- Material codes and standards;
- Decommissioning;

This paper deals with the treatment of the “sodium coolant and cover gas” item.

3. THE ITEM ‘SODIUM COOLANT AND COVER GAS’

3.1. Work asked

To treat this item (as every other items), it was given the following constraints:

- It was necessary to write a small presentation text to present the context of this item and some very general information.
- It was necessary to select a limited number of references considered to cover the whole subject of the item. The number of 150 references was given as an upper limit.
- This study as the whole project must be done in a limited time (from 2000 to mid 2002) and will not continue after. Thus the given list of referenced papers will not have any reevaluation with time.

These constraints and specifically the limited number of referenced papers were the major difficulty to deal with such a wide item. Thus, to try to solve these difficulties, the first operation was to divide this subject into several sub items.

3.2. Description of the sub items

In order to be more precise in the definition of the indispensable knowledge on sodium coolant and cover gas, it was decided to define several sub items. Eight sub items were selected and even in these sub items, another division was made. At the end this strategy leads to the following list:

1) General knowledge on sodium
   - Physical properties;
   - Chemical properties;
   - Thermal properties.

This sub item deals with all the properties of metallic sodium that is necessary to know to have a good knowledge of how the metallic sodium behaves.

2) Primary sodium
   - Specifications for primary sodium;
   - Pollution monitoring;
   - Activation;
   - Measurement techniques;
   - Purification techniques.

The most important operations dealing with the primary sodium are covered by this sub-item.

3) Secondary sodium

- Specifications for secondary sodium;
- Pollution monitoring;
- Measurement techniques;
- Purification techniques;
- Specifications for secondary sodium.

The most important operations dealing with the secondary sodium are covered by this sub-item.

4) Experimental feedback from PHÉNIX and SUPERPHÉNIX

- Recall of the experimental feedback gained from the main incident occurred in PHÉNIX plant in operation;
- Recall of the experimental feedback gained from the main incident occurred in SUPERPHÉNIX plant in operation.

This sub item is covering the major experimental feedback gained by some incidents encountered on French fast operators in operation. That is to say for example several sodium/water reactions on Phénix Steam Generators, pollution of the primary sodium by air ingress on Superphénix reactor, etc…

5) Argon cover gas

- Physical and chemical properties;
- Behavior with sodium – aerosols;
- Specifications for fast reactors;
- Measurement techniques;
- Physical and chemical properties.

All the basic knowledge of the argon cover gas and its behavior in contact with liquid sodium are gathered in this sub item.

6) Narrow items

- Corrosion in sodium;
- Sodium compatibility;
- Sodium potassium alloy (NaK alloy);
- Cleaning – Decontamination – Re-qualification for the reuse of the components.

This sub item covers all the subjects where there was a fear that they will not be properly treated in another item because they are at the borderline between two fields of knowledge.

7) Further or other coolants – Comparison with sodium

It was found necessary to define this sub item in order to remind for the future what were in the past the reasons to select sodium as coolant for fast reactors and not another coolant (such as gas or lead or whatever). And also to remind what were the comparisons between sodium and other coolants with the choice criteria.

8) Books, technical documents and general conferences on sodium technology

Because this subject: sodium coolant, was too wide, and because it was really difficult to select a limited number of document when several thousand have been written on this subject, it was decided to define a sub items where unavoidable referenced documents must be conserved. These document are not concerned to only one sub item but they generally treats all of them. Thus the general international conference on liquid metal technology (LIMET) or some Handbook on liquid sodium was selected in this sub item.

4. DIFFICULTIES IN DOING THIS SELECTION

Several sorts of difficulties arose when time is arrived to select the final papers. These difficulties can be sum up in four questions:

- Am I the best person chosen to select the best document?
- It is of course difficult to identify the right person that knows every about sodium coolant. This question means that even if I have been chosen, my own knowledge is only partial and mainly focused on my field of competence. So, it is necessary to keep in mind that all the field of the subject must be covered and not only what I know the most.
- How can I be sure that I will not miss something important (one paper or worst one part of the field of this item) ?
- How it is possible to select around 150 documents when several thousand were produced all over the world ?
- Of course to solve this problem it should be argued that now we can store everything on computer. There is no more a limitation of space memory. But by selecting everything it is not possible to let to the future generation the experimental feedback of what it is useful to know and what is not (or far less). By storing everything, we let to the future a huge amount of data and documents with no hierarchy between themselves. We are not in a knowledge preservation attitude but in a general archive strategy with no increase in value by French experts. It is not the aim of this project.

To answer as much as possible to these previous questions, the following recommendations happened to be very useful:

- Try to choose only the papers written as synthesis.
- Try to ask to every specialists: “What are the five documents that summarize your field of speciality the best?”
- Try to keep the same importance in every subject and not only in the field of your speciality.
- Try to see abroad if the field of interests is the same or not.
- And finally keep in mind that your selection is maybe not the best but it is better than nothing (!).

5. SYNTHESIS OF THIS STUDY

The selection of all the documents designed to cover this subject is now over. One-hundred-one documents were selected: 92 French documents coming from CEA, EDF and FRAMATOME and 29 foreign documents coming from all the countries that have worked or are still working in fast reactors. Ten general books were selected (sodium handbooks, IAEA conferences on liquid metal technology or specialist meetings, IAEA technical documents). Moreover all the up to date courses given at sodium school were selected because they were easily available and they were a good synthesis of every specific subject (i.e. corrosion, purification, sodium monitoring, cleaning, etc…). The compilation of all the sodium school courses was counted as only one reference but in fact it gathers more than 70 courses.

6. CONCLUSIONS

The list to treat about 'sodium coolant and argon cover gas' is now finished and considered as definitively complete. After doing this kind of work there is still a feeling that nonetheless a lot a knowledge is lost or difficult to maintain through the ages: principally the knowledge of retired experts.

It was possible to notice also that the field of interest in one particular subject in sodium coolant depends strongly on what happens during the LMFRs life. For example for French part, it was mainly sodium pollution and purification consequence of the air ingress in Superphénix and also materials and corrosion as a consequence of the intermediate fuel sodium storage leak. In Great Britain, a lot of documents were focused on sodium/oil interaction due to the oil ingress in PFR. This field of interest is also linked to the questions of the national safety authorities. As an example studies on In Service Inspection and Repair started in a lot of countries in the 90's after questions of safety authorities on this subject. Before, nothing was asked and as a consequence the development in this field was no so important. The trouble is that it is not possible or obvious to predict what will be the field of questions coming from the safety authorities in thirty years.

It can be said as a final conclusion that this selection on sodium coolant is now ready for the future, so the knowledge preservation is on the way. This list of document is also very useful for the present time.

REFERENCES

[1] BAQUE, F., R&D LMFRs Knowledge Preservation French Project, paper presented in the IAEA Technical Meeting on Operational and Decommissioning Experience with Fast Reactors, Cadarache, France, 11–15 March 2002, to be published as IAEA-TECDOC.

JNC VIEWPOINT ON FAST REACTOR KNOWLEDGE PRESERVATION

Y. YOKOTA, P. HARRISON, T. IRIE
Japan Nuclear Cycle Development Institute (JNC), Ibaraki, Japan

Abstract

JNC is undertaking a major program of research and development on liquid-metal cooled fast breeder reactors, which is fully supported by the government of Japan and the electrical utilities. Hence, the perspective of JNC on knowledge preservation is rather different from that of organizations where the fast reactor project has been scaled down or discontinued. Within JNC, there is a statutory obligation to preserve documentary records of the fast reactor project. Over time the method of archiving has changed from optical (microfilm, microfiche etc.) to digital storage. It is the long-term objective of JNC to convert all its records to digital format and make them available to staff over its intranet. JNC is also attempting to preserve 'human knowledge', that is, the expertise of staff who have been involved in the fast reactor project over a long period and who are now nearing retirement. Based on this information, two computerized systems are currently being constructed: one which records in a readily accessible manner the background to key design decisions for the Monju plant; and a second which uses simple relationships between design parameters to aid designers understand the knock-on effects of design choices (joint project with Mitsubishi). To its partners in international cooperation — the US/DoE and the organizations of the Euro-Japan collaboration — JNC is proposing a joint approach to knowledge preservation and retrieval. The proposed concept, dubbed the International Super-Archive Network (ISAN), would make use of the standardized software the new technologies of the internet increase the mutual accessibility of fast reactor information. JNC considers it extremely important to reflect the lessons learnt from previous experience in the fast reactor field to the operation and maintenance of Monju and the design of future reactors.

1. CURRENT STATUS OF JAPAN'S FBR PROJECT

JNC has two operable fast reactors: the experimental-scale reactor Joyo which is currently being upgraded to improve its ability to perform irradiation experiments; and the prototype Monju which remains shutdown since the 1995 sodium leak accident. It is always difficult to predict when Monju will return to operation, but it is hoped that — if all goes well — Monju could be restarted in 2005. Most of the R&D effort in the project is now directed towards the so-called Feasibility Study that is looking at ways to optimize the design of future fast reactors, and the entire cycle, with a view to commercialization. This is a joint project with JAPC and other Japanese research organizations. The point is that — in contrast to what has been happening in Europe and the USA — Japan still has a major fast reactor project that continues to have the full support of the Japanese government and the electricity industry. This gives a very different perspective on knowledge preservation.

2. JNC APPROACH TO KNOWLEDGE PRESERVATION

As a publicly-funded research organization JNC have a statutory obligation to keep all documentary records of the fast reactor project. With such a huge number of documents to manage it is necessary to consider how to make them accessible to JNC's staff. It is the long-term objective to transform all the documents into electronic format and make them available via company intranet. This is an ongoing project. Preserving what could be called 'human' knowledge is also being attempted. It's a real problem for JNC that many very experienced staff is reaching retirement age and are leaving the organization. These were the people who were involved in the design and construction of Monju from the outset, and they have a breadth of experience that would be difficult for someone entering the project today to acquire. Hence, building a computerized system based on their accumulated knowledge is attempted.

2.1. Preservation of human knowledge

There was the key staff, who are now nearing retirement or have already retired, in the Monju design process. Interviews them about their experience are being made. The computerized system, based on this knowledge, is in two parts:

- The first part is a record of the decision making process behind each key design parameter on the plant. Starting with an index of the major systems and components, it is possible to look up, for example: Why was a loop design chosen in favour of a pool? Why three loops?…and so on, down to a more detailed level. Without this system staff would have to search through dozens of old reports to find such an explanation.
- The second part — the design interaction guide — uses simple relationships to illustrate how a change in one design parameter has a knock-on effect on others. This is a joint project with Mitsubishi and it is still at an early stage, but it is hoped to be useful to future designers.

2.2. Retain all documentary records of the FR project

As mentioned, it is JNC's obligation to preserve everything as possible. This includes all the background R&D, the design documents, drawings and so on. It also includes the computer codes that were used at the design stage. A major effort has been made to index all the information by technical area, and already the index is available to JNC's staff over the company intranet. In the future, it is hoped to add a more sophisticated cross-referencing system. There is also a long-term project to select, from this huge archive, a more limited set of key papers. For example, storage media and scale for Monju are as follows (it does not include background R&D or anything related to Joyo). The original paper documents are considered to be the primary reference source, and they are always retained — even after they have been scanned or transformed to some other medium. It is estimated that there are about one million pages of such documents.

A few years ago, the best way to preserve these seemed to be microfilm or microfiche — but this has now been abandoned in favour of digital storage. Documents are scanned into Adobe Acrobat (pdf) format using a high-speed scanner. So far there are about 66 Gigabytes of documents stored in this way.

There are also a lot of photographs. It is policy to photograph all construction and maintenance work that is carried out — in every maintenance team there will be one man with a camera. It is estimated that there are at least 100 000 photos, some slides but mostly prints, and of these about 70% have now been digitized.

Figure 1 shows just one example of situation in the document storage process. In this case the documents considered are the Monju technical materials, design reports, manufacturers reports and so on. The diagram shows roughly how much information is there in each format, but it also illustrates how our strategy has evolved with time. The early materials were copied on to microfiche — but this was replaced by a slightly different optical storage system from Toshiba. Both these have now been superseded by digital computer storage. As shown, there are some materials on paper only, but there is also an overlap where some materials are stored in multiple formats.

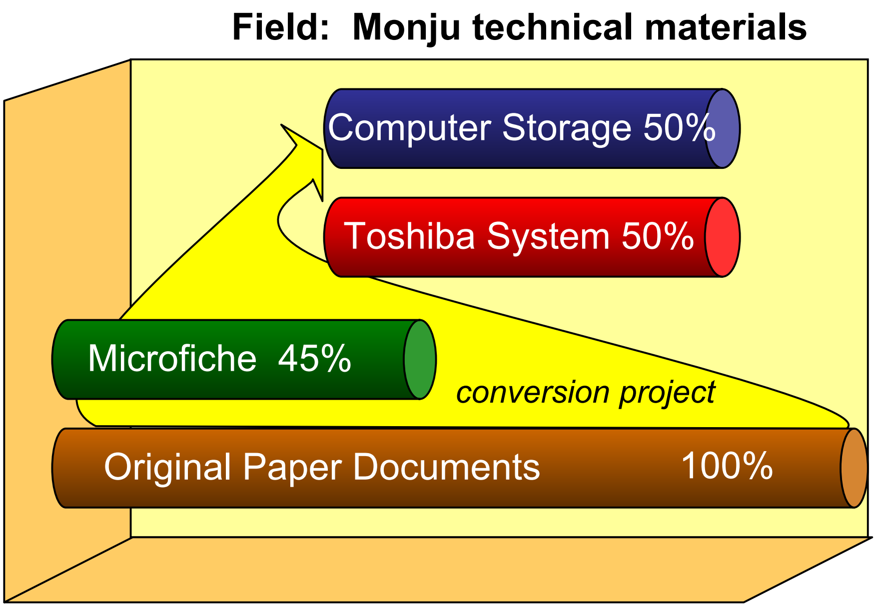


FIG. 1. Example of infromation storage.

3. JNC PROPOSAL ABOUT KNOWLEDGE PRESERVATION

As mentioned, the current status of the Japanese fast reactor project is clearly different from those in our partner organizations in Europe and the USA. There are two operable reactors but relatively little operating experience. By contrast, our partners have a vast amount of valuable operating experience but only one operating plant that is nearing the end of its life.

Over recent years major cutbacks in overseas projects have been witnessed. It is regrettable to see the reduced presence of some countries at international meetings. Many of the staff with whom we used to have contact have now left the fast reactor field to go into other areas or have retired. And this fact would also make it difficult for our partners to locate information.

There are other problems too. Some of the projects closed down quite quickly without much time to consider the problem of knowledge preservation. So the information may exist only as paper documents in the local language. Where knowledge preservation has been undertaken, it has been done so against the background of a project closure, with the belief that this information would not be required for many years to come. So little consideration has been given to accessibility of information to ongoing projects.

JNC has two fundamental issues to address in the future of our international co-operation: knowledge preservation and communication. However, while the fast reactor project has slowed down, progress in computer technology has accelerated. There are opportunities provided by cheap digital storage and by the Internet technology, and it is these that form the basis of JNC's proposals.

3.1. 'International Super-Archive Network' (ISAN)

JNC's possible solution, that is being proposed to our partner organizations, is the concept of an "International Super Archive Network" (ISAN) in which all information is mutually shared between the partners. ISAN is also the Japanese word for "heritage" or "inheritance". The concept is based on the model provided by the worldwide-web. Each organization would build its own archive of documents stored digitally on a server in a popular format such as PDF. Where possible they would do some preliminary categorizing, prioritising key fast reactor fields; but – just like the web – the archives would be largely unstructured and search engines would be relied on, rather than an elaborate hierarchical index, to locate documents. Importantly, this means a lot of experts at the "input end" of the system are not necessary. The onus is shifted from the information provider to the information user. The users know what they want, and it is their responsibility to find it.

The software to do this is all commercially available and will no doubt continue to improve rapidly since it is led by the massive demand of the Internet. Even automated translation can be envisaged. There are two distinct aspects to the ISAN proposal. The first is the important task of ensuring that all documents are put into digital and searchable form. The second is the secure extranet equipped with software to search the archive. These aspects can be treated quite independently and development can be carried out in parallel.

Figure 2 shows the concept of ISAN system with several servers linked by an extranet. The infrastructure provided by the Internet is used, but with encrypted data transmission to form a private network. "Remote worker" in this figure represents an overseas branch office, a university research department, or a manufacturer, which might have a more limited access to the system according to the work that they are involved in. To the user, the front-end of the system would appear like a normal web page.

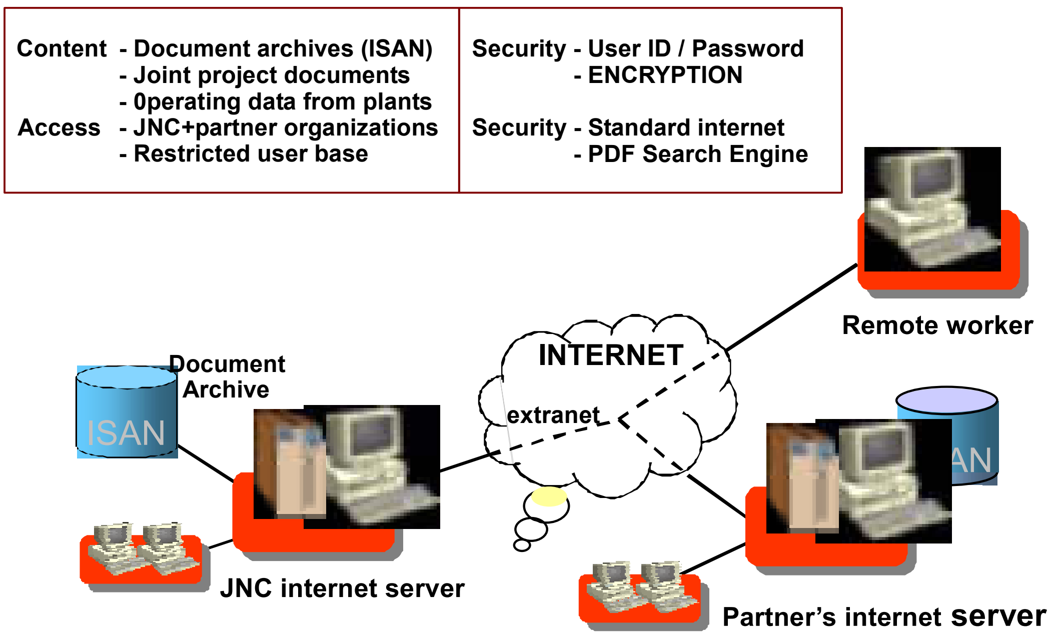


FIG. 2. Concept of extranet and ISAN system.

At the moment this is still only a concept that is being discussed with our partners. There are many issues to be resolved before starting to turn it into a reality, such as data security, and the control of information.

3.2. Indexing & search engines

The classical approach to making documents retrievable from an archive would be to index and sub-index. The user follows the appropriate route to find what he is looking for. In practice there would be parallel routes such as indexing by author. This kind of indexing works well but requires considerable effort to construct, and consequently it is unlikely to cover all available materials.

The search engine uses only keywords to jump directly to the documents. If the search is imprecise this may produce a large number of results; so the responsibility is placed on the user to be more exact in his requirements.

Indexing and search engines are not opposing concepts, they are complementary approaches. If the effort is available to construct an index then the search engine can be configured to search a selected area of the index thus producing a more accurate and refined search. This could be limited to a technical area, particular author, or to key documents according to what has been included in the index.

The quality of search engines has improved greatly in the short time that they have been in popular use and they will no doubt be developed further in future.

3.3. Mutual benefits

There would be mutual benefits for both JNC and its partners in constructing the extranet and ISAN system. Obviously JNC would benefit from the vast experience of fast reactor operation amassed by our partners. But the success of Monju is also important for the future of fast reactor development outside Japan, and the communication would give our partners access to and operational reactor and new experience.

3.4. JNC's international contacts & agreements

Figure 3 shows JNC's international contacts and agreements as viewed from Monju. JNC have multilateral agreements or international organizations that provide a forum for cooperation on fast reactors. There are bilateral agreements in the some fields: fast reactors, advanced technology and so on.

JNC's closest partner in technical matters is CEA. In the first stage of cooperation on knowledge preservation JNC would hope to include EJCC partners and, if possible, the American DOE.

From a technical viewpoint JNC would obviously like to extend the system as widely as possible to include other countries. But JNC is constrained by the structure of these agreements as to the degree of technical cooperation that is possible.

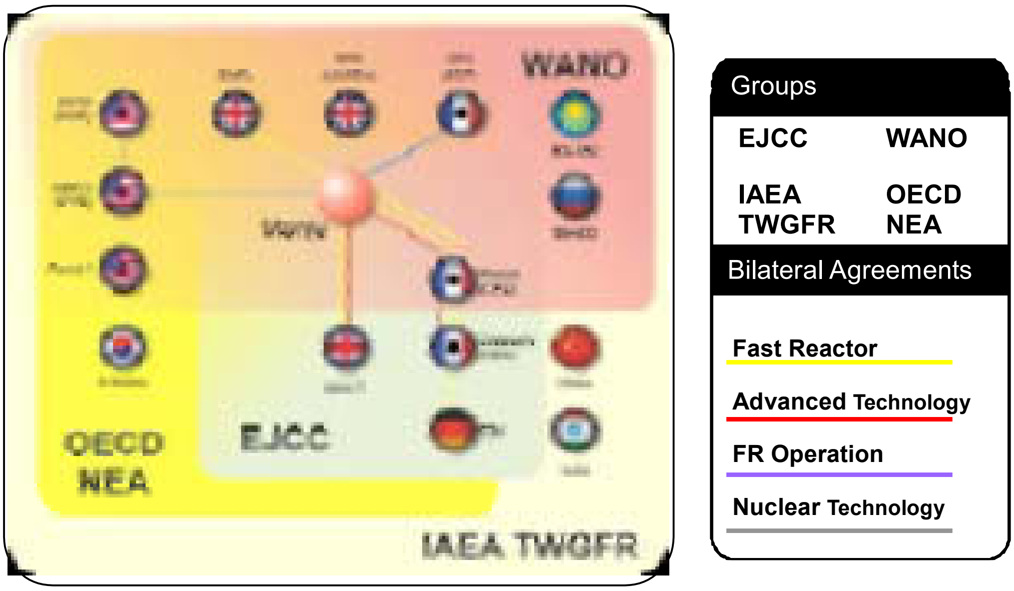


FIG. 3. JNC's International contacts & agreements (as viewed from Monju).

3.5. Special challenges for Japan

There are also some special challenges for Japan in creating the ISAN system. In Japan a mixed writing system is used, which has two phonetic alphabets (hiragana and katakana), each about 50 characters, plus the kanji characters which originated in China — there are over 2000 kanji in regular use — with no spaces between words. This makes the task of optical character recognition a bit more difficult. What makes it more difficult still is that before computers became so widespread, a large proportion of the technical documents were hand-written. At present optical character recognition is almost impossible for these.

Japanese is completely unlike English and the other European languages. The translation software is still not of a high enough standard to be useful.

So, it is difficult, but we believe that this concept could be the way forward for a cooperative project with long-term objectives but limited resources, spread over geographically remote locations.

4. CONCLUSIONS

JNC has the long-term objectives to transform all the documents into electronic format and make them available via company intranet. Further preserving what could be called "human" knowledge is also being attempted.

JNC is proposing a joint approach to knowledge preservation and retrieval. The proposed concept, dubbed the International Super-Archive Network (ISAN), would make use of the standardized software and the Internet technologies.

JNC considers it extremely important to reflect the lessons learnt from previous experience in the fast reactor field to the operation and maintenance of Monju and the design of future reactors.

LIST OF PARTICIPANTS

ASTEGIANO, J-C.	CEA/CE Cadarache, DER/STR Bât 208, 13108 St. Paul lez Durance Cedex, France
ACHOURKO, I.	Institute for Physics and Power Engineering (IPPE), 249033 Obninsk, Kaluga Region, Russian Federation
ALPHONSE, P.	EdF-CIDEN, 35-37, rue Louis Guérin, 69100 Villeurbanne, France
AOYAMA, T.	Experimental Fast Reactor 'Joyo', Japan Nuclear Cycle Development Institute (JNC), Higashi-Ibaraki-gun, Ibaraki 311-1393, Japan
BAQUÉ, F.	CEA/CE Cadarache, DER/STR/LTTS Bât. 201, 13108 St. Paul lez Durance Cedex, France
BERTE. M.	Framatome ANP, 10, rue Juliette Récamie, 69456 Lyon Cedex 06 France
BRACHI, R.	EDF-CIDEN, 35-37, rue Louis Guérin, 69100 Villeurbanne, France
CABRILLAT, M.T.	CEA/CE Cadarache, DER/STR/LTTS Bât. 201, 13108 St. Paul lez Durance Cedex, France
CARBONNIER, J.L.	CEA/CE Cadarache, DER/DIR Bât 707, 13108 St. Paul lez Durance Cedex, France
CHIAROT, D.	EdF-CIDEN, 35-37, rue Louis Guerin, 69100 Villeurbanne, France
GASTALDI, O.	CEA/CE Cadarache, DER/STR/LTTS Bât. 201, 13108 St. Paul lez Durance Cedex, France
GIRAUD, M.	Framatome ANP, 10 rue Juliette Récamier, 69456 Lyon Cedex 06, France
GRABON, V.	Framatome ANP, 10, rue Juliette Récamier, 69456 Lyon Cedex 06, France

HENSLEE, S. P.	Argonne National Laboratory-West, Idaho Falls, Idaho 83403-2528, USA
IBUKI, Y.	JNC (Attache in Phénix), Centrale PHÉNIX, 30200 Bagnols sur Cèze, France
IRIE, T.	JNC (Attache in S-Phénix), S-PHÉNIX, CNPE de Creys-Malville, BP63-38510 Morestel, France
JOULIA, E.	EdF-CODEN, 35–37, rue Louis Guérin, 69100 Villeurbanne, France
LEMOINE, P.	CEA, DEN/DIR (Phenix Project) Bât 121, CEA Saclay 91 191 GIF/YVETTE, France
LOUVET, J.	CEA/CE Cadarache, DER/SERI/LCSI Bât 212, 13108 St. Paul lez Durance Cedex, France
MARTIN, L.	CEA PHENIX, DEN/VRH/DCP/DIR, France
MOUROGOV, A.	EDF/R&D, 1 Avenue du Général de Gaulle, 92141 Clamart Cedex, France
PONCERRY, J.	EDF-CIDEN, 35–37, rue Louis Guérin, 69100 Villeurbanne, France
PFEIFER, W.	Forschungszentrum Karlsruhe GmbH, Techn./administer. Leitung Stilllegung, 76021 Karlsruhe, Germany
RAJAN, M.	Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam – 603 102, Tamil Nadu, India
RAMAKRISHNAN, I.	Indria Gandhi Centre for Atomic Research (IGCAR), Kalpakkam – 603 102, Tamil Nadu, India
RIOU, B.	Framatome ANP, 10 rue Juliette Récamier, 69456 Lyon Cedex 06, France
RODRIGUEZ, G.	CEA/CE Cadarache, DER/STR/LTTS Bât. 201, 13108 St. Paul lez Durance Cedex, France
STANCULESCU, A.	International Atomic Energy Agency, Nuclear Power Division, Wagramer Strasse 5, 1400 Vienna, Austria
SURENDRAN, C.S.	Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam – 603 102, Tamil Nadu, India

SURESH KUMAR, K.V.	India Gandhi Centre for Atomic Research (IGCAR), Kalpakkam – 603 102, Tamil Nadu, India
TOMMASI, J	CEA, CE Cadarache, DER/SPRC/LEPh Bat 230, 13108 St. Paul lez Durance Cedex, France
VINOCHE, O.	CEA/CE Cadarache, DER/STR/LTTS Bât. 201, 13108 St. Paul lez Durance Cedex, France
YU Hong	China Institute of Atomic Energy, Beijing, 102413, China