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Progress in liquid metal fast reactor technology

Proceedings of the 28th meeting of the International Working Group on Fast Reactors held in Vienna, 9–11 May 1995

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FOREWORD

The key objectives and activities of Member State liquid metal fast reactor (LMFR) programmes are:

(1) Demonstration of effective designs,
(2) Demonstration of system safety,
(3) Demonstration of economic competitiveness with other established power generation systems.

The International Working Group on Fast Reactors (IWGFR) at its 1995 meeting observed that while some countries (as a result of static or falling power demand) are reducing the research and development programmes or delaying the commercial deployment of fast reactors, other countries are planning to introduce these reactors and are embarking on their own development programmes. In these circumstances the international exchange of information and experience is of increasing importance.

These proceedings contain updated information from long standing members of the IWGFR and new information on the status of LMFR research and development from new members of the Group: Brazil, China, Republic of Kazakhstan and the Republic of Korea.

CONTENTS

SUMMARY OF THE MEETING

1. INTRODUCTION

The 28th Annual Meeting of the IAEA International Working Group on Fast Reactors (IWGFR) was held in Vienna from 9 to 11 May 1995. The meeting was attended by IWGFR members from China, France, Germany, India, Italy, Japan, Kazakhstan, Republic of Korea, the Russian Federation, the United Kingdom, as well as by an observer from Switzerland.

The objectives of the 28th meeting of the IWGFR were to co-ordinate the exchange of information on the status of fast reactor development (including advanced concepts for radioactive waste burning), to review co-ordinated research activities on fast reactors and to co-ordinate the IWGFR activities with those of other organizations.

2. IWGFR ACTIVITIES DURING 1994

2.1. Technical committee and specialists meetings

A Technical Committee Meeting on Evaluation of Material Coolant Interaction and Material Movement and Relocation in LMFR was held at the O-arai Engineering Center, PNC, Japan, 6-9 June 1994. The participants reviewed and discussed the recent progress in in-pile and out-of-pile experimental data (fuel and coolant interaction) and material movement and relocation during the whole spectrum of core disruptive accidents (CDAs) postulated for sodium cooled fast reactors: (1) sodium boiling and molten fuel-coolant interactions; (2) molten material movement and relocation in fuel bundles; (3) melt penetration and freezing, and (4) molten pool dynamics. The dynamics of material motion and interaction are especially important in the evaluation of core disruptive accidents. This is because the reactivity effects of material motion are crucial in determining the accident sequences and their consequences. The recriticality and resulting energetics potential is primarily determined by the dynamics of material motion. The dynamics of material motion also determine the final material distribution inside the reactor vessel, and this is related to the post-accident heat removal. The meeting covered the entire scope of aspects of material motions which may occur during CDAs. However, this meeting was not restricted to whole-core CDAs but also considered fuel-pin dynamics since this is an important concern in the event of inadvertent control-rod withdrawal. The general consensus of the participants was that the future direction of safety research should also consider some ways to harmonize designs, economical needs and reliable operation of future LMFRs.

A Specialists Meeting on Correlation between material properties and thermohydraulics conditions in LMFRs was held in Aix-en-Provence, France, 22-24 November 1994. The meeting was devoted to thermomechanical aspects of temperature fluctuations in LMFR such as mixing jet phenomena, temperature gradient fluctuations and transfer of fluctuations from the fluid to the structure materials wall. One key objective of the meeting was to identify common trends in the interpretation of experimental and analytical work and the influence on design codes. The meeting noted that great advances have been made in the last few years to improve analytical methods for predicting fluid temperature fluctuation amplitudes and frequencies, involving simulation and direct solution of the Navier-Stokes equation. Within a short time, when improved computing capacity currently planned has become available, it should be possible to provide information on fluid temperature at all times and at all important locations. When this level of detail in thermohydraulic calculations capability is reached, all the requirements for calculating stresses in structural materials will have been met. Two problems are outstanding, however. Firstly, the question of appropriate boundary conditions between more approximate fluid mechanics codes used to predict overall flow properties and codes to predict the fine detail of thermal fluctuation in critical regions has to be solved. Secondly, the adequate experimental validation including measurements of suitable detail and precision has to be provided. Significant progress has been made in substantiating the phenomenological basics of the material design codes available to reactor designers. Nevertheless, in spite of this improvement, the design codes are still very restrictive in terms of allowable temperature amplitudes, because they still include a "safety factor" of about 2. It is clear that progress will be made only when the basis of this safety factor is scrutinized and a firm experimental justification for reducing it is provided.

2.2. Benchmark calculation for a severe accident (unprotected loss of flow, ULOF) in BN-800 reactor

At the Annual Meeting of the IWGFR in May 1994, it was proposed to begin an evaluation of the BN-800 reactor transient characteristics. After discussion with contributions of all member countries a joint IAEA/EC benchmark exercise for a severe accident (ULOF) in a BN-800 with near zero void core has been endorsed by all participants. France, Germany, India, Italy, Japan (PNC and Hitachi), the UK and the Russian Federation have participated in the benchmark exercise. The main aim of the comparative calculation is: (1) to establish a basis for evaluating the pros and cons of a BN-800 type reactor with a near zero sodium void reactivity core design under hypothetical severe accident conditions, as far as an energetic ULOF is concerned, and (2) to analyze the conditions which allow avoidance of prompt fuel and steel melting in the improved fast reactor core under ULOF-type and other severe accident conditions. It was proposed to analyze two cases: (1) fully unprotected loss of flow (all safety rods are out of operation), and (2) unprotected loss of flow (all active safety rods are out of operation but all passive safety rods keep their ability to work normally). The IAEA convened a Consultancy, 5-6 December 1994, to review the general objectives and scope of the comparative calculations, to discuss and agree on the input data and to establish a work plan and work methods for the first year of the exercise.

2.3. Co-ordinated research programmes

A Research Co-ordination Meeting on Intercomparison of LMFR Seismic Analysis Codes, was held from 26 to 28 September 1994, at PNC, Japan, to discuss and compare the experimental and analytical results obtained by various organizations for the analysis of French, Japanese, and Italian tests of Rapsodie, Monju and PEC mock-ups of validating LMFR structural codes. The discussions of the presentations covered a wide range of topics, mostly regarding the core seismic behaviors of the LMFRs. With the present state of the art computer power, it is very difficult to perform an entire core seismic analysis. It was suggested that the central row seismic analysis gives a conservative displacement and shock force. The effect of fuel pins on the natural frequency of the wrapper, which has so far been neglected, should be given a closer look. This may become all the more important in the context of irradiated sub-assemblies, wherein the pins touch the wrapper tube due to swelling. In the context of poor matching of displacement responses for the neutron shield elements between the experiment, the calculations for the Rapsodie data and the need to look into the boundary condition details of the neutron shielding elements were stressed. There is a need for digitized experimental data, without which the comparison of results becomes very difficult.

A Research Co-ordination Meeting on Acoustic Signal Processing for the Detection of Sodium Boiling or Sodium/Water Reaction in LMFR, was held at the Indira Gandhi Centre for Atomic Research, Kalpakkam, India, 1-3 November 1994. The work of this CRP on Development of Diagnostics systems, which can enhance nuclear power plant surveillance, is of interest to NPP operators. The first stage concerned acoustic monitoring of the reactor core by detecting sodium boiling, which produces acoustic effects; the research programme has shown that there is a range of techniques available to detect boiling. The second stage of the CRP covered the acoustic detection of leaks in steam generators units (SGU). The participants were provided with data on background noises from 4 transducers on the SGU, of PFR in UK, leak noise from 4 transducers on the ASB loop in Germany, and mixed files of background and leak noises with various signal-to-noise ratios. The objectives were to: (1) determine the leak's start time and duration; (2) assess the advantage of multichannel analysis over single channel processing for acoustic leak detection; (3) assess the reliability and falsetrip rate of the techniques employed, and (4) evaluate, if possible, the location of the leak. The purpose of this meeting was to: (a) review and discuss the results of investigations carried out by the participating countries; (b) discuss and finalize the preparation of test data from experiments carried out on PFR - SGU for the year 1995, and (c) make recommendations for the future. The RCM has contributed significantly to the progress of the acoustic leak detection in test data made by mixing background noises measured on power plants' SGUs and leave noise measured on loops. The possibility to detecting leaks between 0.1 and 1 g/s in a few seconds was demonstrated.

3. PRESENTATION AND DISCUSSION OF NATIONAL PROGRAMMES ON FAST REACTORS

Presentations on fast reactor development were made by the Members of the Group as well as by the observer. A short review of the discussion is presented below:

France: Due to a lower electricity growth rate and the improved fuel utilization in modern nuclear power plants, the commercial introduction of fast breeder reactors is being postponed. Meanwhile, application of an additional important aspect of LMFR - to transmute long-lived nuclear waste and plutonium burning - is being developed. The current programmes on operation of the existing fast reactors Phenix and Superphenix (SPX) and development of the new one, thus reflect these requirements. At the end of December 1994, the Phenix reactor was approved to work at a power level of 350 MWth. One of the objectives of extending the lifetime of the reactor by additional 10 years is to perform the necessary irradiation experiments to support the CAPRA (Consommation Accrue de Plutonium dans les Rapides) project. The CAPRA programme, initiated in 1993, aimed at demonstrating the feasibility of burning plutonium at highest rates in a fast reactor. An exploratory investigation was made in a study of uranium - free cores allowing the highest plutonium burning rates, i.e., about 700 kgPu/GWe year. The work on the CAPRA programme is being performed in the framework of the European R&D collaboration on the

European Fast Reactor (EFR) Project, and in close cooperation with the EFR Associates. The EFR project has now reached an important milestone with the completion of the concept validation phase. The twin goals for the design, that the customer utilities set at the outset have been demonstrated as achievable: (1) economic performance with the generating cost of the commercial series of EFR is competitive with contemporary PWRs, and (2) EFR is licensable in all participating countries with a safety level requirement which meets the ambitious goals of future nuclear plants. The R&D support has been extensive and has provided comprehensive validation of the design features necessary to meet these goals. The R&D is planned within the context of specific international collaboration agreements existing between Europe and Japan and between France and Russia (CEA-MINATOM agreement where Germany and the UK are "Associated Partners"). The next period will seek to maximize the extent and benefit of collaboration within Europe, to include parties with a strategic interest, particularly in the full fuel cycle, and outside Europe and to realize the potential which has been established, for extending collaboration to the USA. All this should be seen in the context of a decision (to be made around 2005), to start the construction of a fast reactor. The SPX reactor was granted a routine regulatory permission to restart and operate by the Minister of Industry and Environment in 1994, in accordance with advice of the Nuclear Safety Agency after a final safety review. The report of the safety authorities declared that the safety level of SPX was the same as that of the 54 PWR reactors of the French nuclear park. Criticality was reached in August 1994 and the reactor's power was increased in successive steps. The knowledge acquisition programme, which has been developed in 1994, necessitating the operation of the SPX plant, has three major complementary objectives: (1) to demonstrate the capacity of a fast reactor in order to produce electricity on an industrial scale, while contributing to plutonium management and the reduction of long-lived radioactive waste; (2) to study the flexibility of a fast reactor, using plutonium as fuel and to qualify the technical solutions developed within the framework of a research programme, with the main focus on operating this type of reactor as a plutonium consumer, and (3) to study the possibilities of destroying long-lived radioactive waste, in particular the minor actinides, americium and neptunium, within the framework of the SPIN programme and fuel performance (the CAPRA mixed oxide fuel has high Pu enrichment, up to 45%, and would be irradiated to high burnup).

Germany: The fast reactor activities of Forschungszentrum Karlsruhe (FZK) are part of the Nuclear Safety Research Project. The R&D programme of this project has been restructured in accordance with the demands of the Federal Government. The key issues and tasks of the programme concern LWR and LMFR safety and transmutation of minor actinides. The activities have been concentrated in two areas: (1) the participation in the European CAPRA programme (neutron-physical and reactor safety works), and (2) chemical studies for the separation of actinides from lantanides of solvent extraction. Analyses of an unprotected loss-of-flow for the EFR core design have been repeated in 1994 using the improved SAS4A system. Results have shown that consequences of the accident initiation phase are significantly reduced, compared to those calculated in the previous analyses using FRAX-5D, PHYSURAC and earlier SAS4A versions. Differences are mainly due to the improvements in the simulation of the fuel pin behaviour during steady-state power operation (up to high burnup levels), and in particular in the simulation of the post-failure materials relocation. A great variety of fast breeder fuel pins have been tested under operational transient conditions in sodium-cooled capsules in the High Flux Reactor (HFR) in Petten/Netherlands. The test field included power ramping, overpower, power-to-melt and temperature transient experiments and the irradiation was terminated at end of 1993. In 1994, post-irradiation examinations and evaluation work was completed, leading to interesting results: (1) in the test KAKADU 30 an overenriched fuel pellet within a normal fuel stack (power peaking factor 2.25) was irradiated for a few days; the overpower of about 900 W/cm in the"hot spot" fuel pellet did not cause overheating; (2) in the test POTOM-4 three fuel pins containing 15, 20 and 30% of Pu were preirradiated under nominal power conditions of about 500 W/cm and then by shuffling to a higher neutron flux position attained about 800 W/cm. In the neutrographs of all 3 pins fuel centre melting is clearly to be seen; and (3) in one specially designed KAKADU capsule 2 MOX fuel pins (preirradiated to 9 % burnup in PHENIX) were irradiated at 450 W/cm nominal conditions during a cycle of 26 days and then brought to the maximum power of 720 W/cm, i.e. a power ramp of 160% of nominal power with a power rise of 1% per second. Despite the high burnup of 9 at% and a neutron dose of about 70 displacements per atom (dpa), the fuel pins remained intact, showing a slight increase in diameter, but only up to 50 microns. In the frame of the project CAPRA, irradiation experiments are planned for the study of the transmutation of actinides and long-lived products; and the behaviour of new fuel types, which are defined by either a very high content of plutonium (> 40 at%), uranium and minor actinides (MA), or by the complete absence of uranium (in order to avoid additional breeding). The irradiation experiment TRABANT (Transmutation and Burning of Actinides in TRIOX) was planned and executed in a tri-lateral cooperation with CEA and FZK. The Institute's activities concentrate on the fabrication of fuel pins to be irradiated in a TRIOX capsule, in the HFR reactor at Petten. The purpose of the irradiation is to study the behaviour of the new materials under neutron irradiation, in order to get information on such parameters as Pu- and MA-distribution, separation processes and solubility behaviour in nitric acid.

Italy: The key issues and tasks in nuclear R&D in Italy are focused on the following areas: (1) collaboration with General Electric on ALMR (optimization of safety parameters for a fast burner oxide core, studies with the aim of reaching the best burning capability of the PRISM MOD B oxide core, satisfying the safety criteria); (2) seismic isolation systems (guidelines development, seismic isolation tests, analytical studies), and (3) SPX reactor commissioning and service activities.

United Kingdom: The funding provided by the nuclear industry in 1995 will maintain UK's contribution to fast reactor development in the following areas: (1) participation in the activities of the European Fast Reactor Utilities Group (EFRUG) and in strategic fuel cycle studies; (2) engineering support to the EFR reactor design studies in the areas of core physics, in-service inspection and repair (ISIR), containment, decay heat removal and feedback of PFR experience, and (3) UK involvement in the EFR R&D studies, particularly on the CAPRA project in the areas of core physics, safety and fuel performance. The only remaining UK Government-funded activity which contributes to the European co-operation is the PFR closure experiments on sodium reactions and materials examinations. As was reported in 1994, operation of the PFR ceased as scheduled at the end of March 1994, following a very successful final year of operation. At that time, the reactor was handed over to UKAEA-Government Division to begin decommissioning. Reprocessing of the PFR fuel has continued. A total of 10 core assemblies, 19 radial breeders and 456 kg of U + Pu fuel fabrication residues were processed in the 1994/1995 fiscal year. In total, 4048 kg of U + Pu (327 kg Pu) was processed. The closure of the reactor and its subsequent decommissioning provided the rare opportunity to conduct various tests and to obtain information of considerable value. AEA Technology, with the support of industry partners in UK and EFR in Europe, drew up a number of proposals for work to be undertaken around the time of reactor closure. The main of these proposals was to be of value to future fast reactor development and, in particular, to the future plans of the European partners and for the support of the continued operation of the French fast reactors, particularly SPX. The programme also had the support, in principle, of the Department of Energy, although the Department advised that it could only financially support a significantly smaller programme than had been proposed initially. The final programme has three parts: (1) steam generator leak detection studies in PFR; (2) sodium-water reaction tests in the Super Noah rig, and (3) destructive examination of materials from PFR. The first two studies took place in the 1993/94 and 1994/95 fiscal years and were completed on 31 March 1995. The materials examination programme began in the 1994/1995 fiscal year and will be completed by the end of March 1996.

Switzerland: Within the framework of the CAPRA project, the fuel option for amplified plutonium consumption is being studied. In the area of materials for actinide transmutation, the following tasks has been completed in 1994: (1) preparatory experiments and solubility tests for $(U_{1-x}\ PU_x)\ O_2$ (0.25 < x < 0.65 and for $(U_{1-x}\ PU_x)$ N (0.25 < x< 0.75), as possible materials for the efficient fission of plutonium in a fast neutron flux; (2) fabrication of pure PuN-microspheres for ceramic-metal fuel; (3) design calculations for sphere-pac segments, based on the idea of a ceramic-metal fuel; (4) material preparation of (U, Zr) N and pelletization tests of TiN and (U, Zr)N for the irradiation experiment in the reactor PHENIX; (5) experimental preparations of $(Ce,U)\ O_2$, $(Ce,U,Pu)O_2$ and $(Ce,\ PU)O_2$ for the CAPRA core with lower Pu content, and (6) cleaning of americium from waste streams of the plutonium separation equipment (extraction chromatography).

Japan: Japan Atomic Energy Commission formulated a new "Long-Term Programme for Development and Utilization of Nuclear Energy" in June 1994. Pu utilization is a key in the development for long-term energy supply and energy security of Japan. Therefore, Japan intends to steadily and systematically pursue the development of fast breeder reactors. The status of FBR development in Japan in May 1995 was as follows: (1) experimental fast reactor "JOYO": 51,100 hours operation since 1977, (2) the 280 MWe Prototype FBR "MONJU": start-up tests are continued; (3) the 660 MWe Demonstration Fast Breeder Reactor (DFBR): a three years design study for plant optimization, started in 1994 by the Japan Atomic Power Company (JAPC), and (4) the FBR fuel cycle: the MOX fuel fabrication at the Pu fuel production facilities for Joyo, Fugen, BWR and Monju is in operation. The R&Ds for FBR fuel recycling are underway at three demonstration facilities. The construction of one of them was initiated in January 1995.

Russian Federation: There are certain specific features of fast reactor development in Russian Federation. Russian Federation has about 100 reactor-years of positive experience of operation of several experimental and one semi-commercial fast reactor, the BN-600 with a power output of 600 MWe. In 1994 the load factor (LF) of the BN-600 reactor was 78.2% (80.3% in 1993). This reduction of the LF compared to one in 1993 was caused by increase of heat output for consumers (0.49%), longer-planned repair works and fuel refuelling time (0,66%) and electrical grid limitation (0.53%). The average LF for the period 1980-1994 equals 70%. In 1994, an incident took place at the secondary sodium circuit. During the planned outage (replacing of the valve in the draining system) in May 1994, about 1.1 tonnes of non-radioactive sodium had leaked from secondary loop through the 48 mm diameter drain pipe, but only several tens of kilograms were burned. The remaining sodium was retained in the flow-smothering cath pan system covered with the extinguishing power. By the end of 1994, the BN-600 reactor total on-power operation time amounted to 102, 139h and 53,100 GWh electricity was generated. Cost of electricity produced by BN-600 in 1994 was approximately 20% lower than that for fossil-fuel plants operating in this region. Experience gained from the 15 years of commercial operation of the BN-600, and new advanced projects (BN-600M, BN-800), indicate a potential for competitive energy production by fast reactors. Reprocessing plants for uranium and power plutonium extraction from spent fuel of LWRs and LMFRs are also operating in the country. The rate of introduction of fast reactors is lower compared to earlier forecasts, and a closed MOX fuel cycle has not yet been realized for the light water reactors. Conversely, there is an accumulation of Pu from the civilian programmes (as much as 30 tons) and dismantled nuclear warheads (about 100 tons). Disposition of this large amount of Pu is of serious concern. Many Russian experts believe that in their country, the utilization of Pu in fast reactors is more feasible than in light water reactors. Two cores of weapon grade plutonium have been tested in the BR-10 experimental fast reactor. For many years, the BOP-60 research reactor has been operated by recycling its own Pu. In the former USSR, BN-350 reactor tests have been performed with subsequent investigation and reprocessing of fuel subassemblies with MOX fuel (350 kg of weapons grade Pu). Taking into account the above mentioned factors and considering the worked out concept of nuclear power development until 2010 for the next stage of Russian fast reactor development, the construction of 3-4 BN-800 plants is planned. The main objectives of R&D for the next generation of LMFRs include: (1) development of advanced fast reactor technology to reduce the capital costs and construction time; (2) certain improvements in safety: passive reactivity shutdown and decay heat removal systems, core catcher and improved containment, and (3) development of new core arrangements for weapon-grade Pu burning, and a development of irradiated fuel-reprocessing technology from the standpoint of deeper extraction of minor actinides in order to arrange their burnup.

India: Major activities within the Indian LMFR programme are the operation of the Fast Breeder Test Reactor (FBTR) and the design of the Prototype Fast Breeder Reactor (PFBR). The 40 MWth FBTR provides a basis for engineering and physics experiments. In 1994, the mixed carbide fuel achieved a peak burnup of 16,000 MWd/t without failure. The sodium pumps and other equipment are operating very well and have logged more than 66,000h. Preliminary design of the 500 MWe, pool-type, sodium-cooled PFBR was carried out in 1985. Considerable expertise has been developed during the last years in analytical capabilities in the field of reactor physics & shielding, thermal hydraulics, structural mechanics, and safety analysis. Experimental facilities for thermal hydraulics and structural mechanics studies, testing of components in sodium and development of special instrumentation for sodium systems are being set up. R&D in fast reactor technology area, including fuel reprocessing, safety research and health physics are continuing at Indira Gandhi Centre for Atomic Research (IGCAR). About 1500 scientific and technical staff continue to work on the FBR programme with a budget provision ~ US$ 30 million for 1994-95.

China: The completion of the construction and initial criticality of the 65 MWth (25 MWe) Chinese Experimental Fast reactor (CEFR-25) in the year 2000 have been approved by the Chinese Government, as one of the targets of advanced reactor development in China. Based on the validation and some optimization of the conceptual design, the technical design of the CEFR was started in early 1995, with some special requirements as to reactor safety and environmental impact. Concerning the research and development of fast reactor technology, progress in sodium technology, materials, fuels, sodium instrumentations etc., has been gained.

Korea, Republic of: The objective of the fast reactor R&D programme that was started in Korea, is to establish an advanced LMFR design concept in order to contribute to a certain amount of worldwide LMFR R&D works and ultimately to the LMFR industry as a whole. Experiences in LMFR technology include a joint feasibility study of Korea and France (1984-1987) on the introduction of a commercial FBR Power Plant in Korea and a joint pre-design concept study with KAERI and General Electric (GE) (1995-1997) on a Korean LMFR with comparisons of the GE and Japanese concepts. The goals of the Korean LMFR development plan are as follows: completion of the basic design of the Korean prototype liquid metal reactor LMFR (KALIMER) with power between 150 and 350 MWe by 2001; detailed design - 2005, and first criticality - 2011. The commercial LMFR project for 2025 should start around 2015.

Kazakhstan: The BN-350 LMFR (on the shore of the Caspian Sea, Manguslak Peninsula) is the first nuclear reactor in the world producing heat for seawater desalination. The reactor that was designed for an estimated lifetime of 20 years has been used for seawater desalination and electricity production for 23 years, the longest of any commercial LMFR. Since the reactor power is used to produce desalinated water (80000 tonnes/day) and to generate electricity (130 MWe) for the region, the plant operator intends to extend the plant life as long as safety conditions permit. The reactor life extension is under detailed evaluation and preliminary results seem favourable, since the plant has been operated entire time at 50-70% of nominal power. There is a detailed plan of safety upgrading related to new requirements. However, their implementation has shown some delays due to financial difficulties.

Brazil: Research activities on fast reactors are being (or are planned to be) executed in the following areas: a reference design of a 60 MWth experimental LMFR is completed, based on some parameters and on general description of LMFRs already built in other countries; U-Zr alloys are being studied on a laboratory scale, with the objective of learning how to fabricate and characterize them; metal fuel recycling is being researched, using electro-refining technique. A first sample of actinide electro-deposited fuel is expected by the end of 1995; a laboratory scale electromagnetic pump has been developed and tested, using mercury as the working fluid. The following institutes are involved in the LMFR activities: Nuclear Research Institute, Technological Research Institute and Aeronautic Technological Institute.

European Commission: The European Commission continued its fast reactor related activities in 1994 along the same lines as in the past, in a reduced fashion, with emphasis on applications to actinide and fission products transmutation. Safety research is performed at the Safety Technology Institute (STI) of the Ispra Establishment and Fuel Research at the Transuranium Institute (TUI) of the Karlsruhe establishment. Investigations of operational limits for future fuels were continued in 1994. The high burn-up mixed nitrides and technetium irradiation experiment POMPEI (POM Petten Irradiation) in HFR-Petten has achieved the goal burn-up (27.8 $^{a}/_{0}$) and was unloaded from the reactor for post irradiation examination. The high burn-up irradiation experiment POMPEI, containing targets loaded with mixed nitride fuel and technetium, has reached its goal burn-up after 270 days of irradiation in HFR Petten. The goal of this experiment is to obtain data on the evolution of structure of fission products and their chemical behaviour at high burn-up. The final results of the MOL 7C /6 and /7 tests, which were organized as a shared cost action of the reactor safety programme in the area of LMFR safety, have been discussed in 1994 and documented in two final reports dated March 1995. This programme focused in particular on the behaviour of a LMFR core under local flow coastdown accident conditions and consisted of two actions, namely: (1) the in-pile study of the propagation potential of local subassembly faults, with emphasis on clad failure monitoring, damage evaluation and post-fault inherent coolability (action 1 -local subassembly faults / "MOL 7C Programme") with the final Report "In-pile local blockage experiments MOL 7C/6 and /7", and (2) the fission product inventory and the radiological impact of irradiated pin failures on the environment (action 2 - source term measurements / "Measurement of fission products in the experiments MOL 7C/6 and 7)" with the final report "MOL 7 C/6 and 7/ Measurement of fission products".

CONCLUSIONS

The IWGFR observed that while some countries are reducing their research and development programmes and delaying the commercial deployment of fast reactors, an increasing number of other countries are planning to introduce fast reactors and embark on their own development programmes. Given these circumstances, the international exchange of information and experience is of an increasing importance to ensure safe practice in LMFR design and operation, by recognizing that no country is successful in developing such technology in isolation. The Group therefore concluded that it was important that the activities of the IWGFR continue, to provide the unique forum in the world for such international exchange.

THE STATUS OF FAST REACTOR TECHNOLOGY DEVELOPMENT IN CHINA

XU MI
China Institute of Atomic Energy,
Bejing, China

Abstract

Delayed almost 20 months, it is still waiting to win the approval of the State Planning Commission for the Chinese Experimental Fast Reactor (CEFR) project. Some meetings to demonstrate the necessity to execute now the program of the CEFR design and construction have been conducted in China. Fortunately the possitive conclusion has been obtained, which would be an important promotion to win the approval.

Based on the validation and some optimization of the conceptual design, the preliminary or technical design of the CEFR has been started in the early of 1995 with some special requirements on reactor safety and impact to environment .

Concerning the research and development of FBR technology, some progresses on sodium technology, materials, fuels, sodium instrumentations etc. have been gained and briefly described in this paper.

1. Introduction

As one of targets of the Advanced Reactor development in China, the construction completion and initial criticality of the 65 MWth (25 MWe) Chinese Experimental Fast Reactor (CEFR-25) in the year 2000 has been approved by the Government on March 14, 1992.

After the review of the preliminary feasibility study reports including the General Report, Conceptual Design of the Reactor, Site Evaluation Report, Safety Evaluation, Environmental Impact Evaluation, and Economic Analysis was completed in March, 1993 by the China National Nuclear Corporation (CNNC) in the name of the Government Authorities, then, starting to compile the feasibility study reports with the same titles as the preliminary feasibility study reports, we have been waiting since May, 1993 the approval of the project by the State Planning Commission to arrange it as annual project which means the project could be financed annually.

The approval has been delayed up to now, mainly due to the investment hasn't been settled and more or less it has been inferenced by some negative news on governmental policies to fast reactors from other countries. Actually, the importance of fast reactors to the energy resources development in China is never changed.

China has become one of the countries with a very fast growth of electricity production. The capacity increase in 1993 and 1994 were 14.38 GWe and 15.0 GWe, the total electricity capacity were 180 GWe and 195 GWe and the total amount of electricity generation were 815 TWh and 883 TWh respectively. But the electricity generation is still far from meeting the demands for development of the national economy and for an improvement in the standard of living.

China is a country with a vast territory. Its conventional energy resources are numerous, but they are not evenly distributcd and the average amount per capita is low. About 60% of the coal reserves are concentrated in north and north-west China and about 70% of hydropower potential are scattered in south-west China, whereas in the coastal areas with both a dense population and a well developed economy the energy resources are extremely scarce. For transportation of energy source, almost 48% of the railway capacity and 25% of the motorway capacity are needed which results in unreasonbly high price of the energy. For example, the coal price after transportation to Shanghai is a few times of that in Shanxi Province and even more while it arrives to Guangzhou city. Both transportation of energy source and electricity generation appear to be the bottle-necks of the development of the national economy, especially in the south-eastern coastal areas. Obviously, to surmount those bottle necks, we have few option other than development of nuclear power.

The necessity of developing nuclear power should also be considered from an environment protection and carbon-hydrogen chemical industry resurces preservation point of view.

In the Chinese nuclear power programme, PWR has been selected for the first generation of reactor types. Based on it the lack of Uranium resources will be occurring if the total nuclear capacity in large scale is needed. So the FBRs will play a very important role for meeting the nuclear programme in large scale to replace coal-fired power plants as more as possible.

Recently some meetings for proving the necessity to design and construct now the experimental fast reactor have been held in China. The conclusion drawn from those meetings is positive and conducive to the approval for the CEFR by the State Planning Commission.

2. CEFR Design

2.1 Status of CEFR Design

After three years preparation for the CEFR design, including the development, collection and review of about 50 computer codes and the decision of main technical selections and of design boundary conditions, from 1990 to July 1992, the conceptual design of the CEFR has been completed. In March 1993, based on the conceptual design, the preliminary feasibility study reports were approved by the China National Nuclear Corporation in the name of the State Authorities. The confirmation and some optimization of the conceptual design have been carried out from October 1992 to the end of 1993, the main design results have been briefly given at last IWGFR meeting.

Spent almost whole 1994 for its preparation, the CEFR technical design has been started in the early of this year.

2.2 Some Safety Requirements to CEFR

Based on the CEFR conceptual design the input for the CEFR technical design has been settled in 1994, in which some safety requirements to the reactor should be mentioned as following.

2.2.1 Passive Safety Properties

As main technical selections, it has been proposed that the CEFR should be provided with passive safety properties, in other words, during any credible transient incident for example LOFWS, LOHSWS, TOPWS, etc., by the negative feedback of reactivity the reactor will enter and be keeping at safety condition, without needs of any personal interference. The residual heat will be removed away by natual convection and natual circulation.

In the conceptual design stage two same passive Residual Heat Removal Systms (RHRS) have been conceptually designed, each one composed of a heat exchanger immersed in hot pool and two air coolers with their own chimney. Two computer codes are under developed in Qinghua University and Shanghai Jiaotong University respectively. To prove them, a reactor pool water Mockup in 1/5 scale of the CEFR has been fabricated (photo 1). Another water mockup based on the CEFR technical design is planned for demonstration of natural convection capability.

The favourable condition to realize the passive RHRS exists in CEFR, because it has more quantily of primary sodium relative to its power as shown in Table 1. It means that it will be easier to find a position in h ot pool for the heat exchanger of RHRS to realize the natual convection.

Table 1 Specific Primary Na Quantity

Reactor	Power(MWt)	Primary Na(t)	Ratio(t/MWth)
EBR-II	62.5	286	4.58
Phenix	563	800	1.42
PFBR	1210	1200	0.99
PFR	670	850	1.27
BN-600	1470	770	0.52
SPX-1	3000	3300	1.10
BN-800	2100	820	0.39
EFR	3600	2200	0.61
CEFR	65	260	4.00

2.2.2 Requirements to Radioactive Materials Releases from CEFR

2.2.2.1 Operational States (Normal Operation and Anticipated Operational Occurrences)

The public maximum annual effective dose equivalent as additional exposure due to releases from nuclear plant of gases and liquids containing radioactivity as a consequence of operational states, specified by Chinese National Authority is not more than 0.25 mSv (GB 6249-86). In the case of

CIAE, all old facilities containing radioactivity have offered 0.07 mSv as maximum exposure to the public. So 0.05 mSv is given as the limit of public annual effective dose equivalent from the CEFR and rest 0.13 mSv for future facilities.

Under operational states the limits of the annual releases from the CEFR of radioactive gases and liquids are given in Table 2 in which calculation values are based on the CEFR conceptual design.

Tabel 2 Limits of Radioactivity Releases

Gases	Calculation(Bq/a)	limits (Bq/a)
Inert gas	5.99×10^{12}	5×10^{14}
Iodine	2.61×10^{8}	1.5×10^{10}
Particles	2.4×10^{7}	4×10^{10}
Liquids		
Tritium	7.7×10^{7}	3×10^{13}
Other nuclides	2.5×10^{9}	1.5×10^{11}

2.2.2.2 Accident States

2.2.2.2.1 Design Basis Accidents (DBA)

According to the guidance HAF 0703 issued by the State Nuclear Safety Administration, the intervention level is 5-50 mSv to whole body for hiding oneself and 50-500 mSv to whole body for evacuating when the emergency of nuclear radiation accident has happened. Furthermore the CEFR site is only about 45km far away from Beijing city which has more than 10 million residents. So hope that no any emergency action for any accident happened.

Therefore, it is stipulated that after a Design Basis Accident the maximum public effective dose equivalent should be less than 0.5 mSv (for thyroid 5 mSv) in the CEFR case.

2.2.2.2.2 Beyond Design Basis Accidents (BDBA)

As the same reasons for DBA, after a BDBA the maximum public effective dose equivalent should be less than 5 mSv (for thyroid 50 mSv).

In the accident duration (30 days) the acceptable collection effective dose of residents in the region of 80km radius should be less than 2×10^{4} man Sv, the same value also for collection thyroid effective dose.

2.2.2.3 Intervention Requirements

The sufficient measures of reactor safety control and of prevention from radioative material release should be provided in the CEFR so that there is no any emergency intervention requirement for the residents beyond 400 m from the reactor site.

3. Research and Development on FBR Technology

3.1 Multigroup Constant Development

A multigroup constant library CL5OG which means Chinese Nuclear Data Center (CNDC) library of 50 groups. It is generated through adjusting L50G library which was generated by NJOY-89.31 from ENDF/B-6 and JEF-1 at CNDC. The CL50G library includes 52 nuclides and will be used for fast reactor design calculations in China. Its energy structure is the same as those of LIB-Ⅳ 50 group Library given by LANL of the United State of America.

The adjusting above-mentioned or the difference between CL50G and L50G is as follows:

The structure material nuclides, for example, iron, chromium and nickel have a complicated resonance structures from 0.3 to 5 MeV, but there are no resonance parameters in ENDF/B-6, JEF-1, JENDL-3 and CENDL -2. Consequently self-shielding effects in high energy region for these structure material nuclides can not be calculated using the PASC-1 code system which was developed at CNDC for calculating critical fast benchmark assemblies. In other words, the scattering cross section for these nuclides in high energy region are overestimated. Therefore the L50G library should be adjusted. In

Table 3 K_{eff} values for fastbench mark assemblies

Assembly	CNDC				Caro		JNDC△
	L50G	CL50G	B-4	B-6	JEF-1	JEF-2	JENDL-3
JEZEBEL-23	0.99420	0.99349		0.9929	0.9920	0.9756	1.0206
FLATTOP-23	1.00690	0.99307		1.0026	0.9766	0.9836	1.0175
THOR	1.00484	1.00603		1.0056	0.9923	0.9797	0.9985
JEZEBEL	0.99765	0.99944		0.9960	1.0095	0.9952	1.0001
JEZEBEL-Pu	0.99863	0.99999		0.9893	1.0024	0.9898	0.9963
FLATTOP-Pu	0.99326	0.99020		1.0025	1.0054	0.9887	0.9974
GODIVA	0.99979	0.99908		0.9954	0.9995	0.9934	1.0066
FLATTOP-25	0.99909	0.99497		1.0007	0.9984	0.9898	1.0033
BIG-10*	1.00895	0.99601		1.0063	1.0016	0.9928	1.0038
						JENDL-3△*	
						J3T	J3TR1
ZPR-3-6F	1.01094	0.99900	1.01389			1.02644	0.99855
ZPR-3-12	1.01630	1.00155	1.01467			1.01451	0.99291
ZPR-3-11	1.01283	0.99871	1.00682			1.01271	0.99369
ZEBRA-2	1.01765	1.00177	0.99367			0.99777	0.97657
ZPR-6-6A	1.02015	0.99706	0.99078			1.00916	0.98626
SNEAK-7A	1.01470	1.00356	0.99967			0.99581	0.99750
ZPR-3-54	1.05283	1.00302	0.96200**			0.96287	0.96460
SNEAK-7B	1.01208	0.99672	0.99570			0.99201	0.99520
ZPR-3-50	1.01564	1.00403	0.98982			0.99584	0.99865
ZPR-3-48	1.01601	1.00107	0.99897			0.99636	1.00027
ZPR-3-49	1.01652	1.00102	1.00110			0.99633	1.00066
ZPR-3-56B	1.02137	1.00392	0.98506			0 99687	0.99876
ZPR-9-31*		0 99808					
ZPR-6-7	1.01608	0.99615	0.98697			0.99306	0.99965

* The measured value for K_{eff} of BIG-10 is equal to 0.996 ± 0.001. All of others are 1.0.

** Quoted from Nuc Sci. and Eng., Vol. 57, No. 3, July 1975.

△ Quoted from JAERI-M 91-032, p. 148, March 1991.

△* Quoted from JAERI-M 89-026, p. 90, March 1989.

order to avoid over the limitation of adjustment, the multigroup data have been adjusted within the range given by several evaluated microscopic cross sections. The elastic scattering cross sections above 0.82 MeV for iron, chromium and nickel are decreased by 5%. The γ values, fission and capture cross sections for U-235, U-238 and Pu -239 are a little modified. compared with others, the calculated results with CL50G are better. As an example, the comparison of keff values is given in Table 3.

3.2 Hydraulics

A computer program for calculating fluid hammer phenomena in sodium and water circults of the CEFR has been prepared by Qinghua University. Due to the pool hydraulic properties, there is no any fluid hammer phenomenon at some transient conditions (Table 4)

Table 4 Presure at different section in primary circult mNa

Position	Normal Operation	Primary Pumps Loss of power	Primary Pumps stuck
Pump inlet	7.50	7.50	7.50
Pump outlet	37.69	37.69	37.69
Core inlet	37.24	37.24	37.24

The fluid hammer phenomena are also calculated for secondary circult at some transient conditions, which results are presented in Table 5.

Based on the VOF method (Volume of Fluid) proposed by U.S. LANL, another computer code for calculating the free Na surface wave motion in the CEFR primary sodium pool has been developed in Qinghua University .

By the computer code, the CEFR pool sodium wave motion evolution and maximum and minimum wave level and the wave height have been calculated and given in Fig.1-8 with the conditions besides the geometry parameters:

sodium speed in core 5.7 m/s

cover gas presure 0.5 bar

Table 5 Presure at different section in secondary circult mNa

Position	Normal operation	Secondary pumps loss of power Valve shut in 2 s	Seconday pumps Loss of power Valve shut in 3 s
Pump inlet	28.00	31.69	42.93
Pump outlet	63.26	63.26	63.26
IHX inlet	53.66	108.56	101.05
IHX outlet	48.38	108.63	101.09
SH inlet	46.90	53.88	52.46
SH outlet	39.08	54.59	52.46
EV inlet	38.36	56.84	54.61
EV outlet	31.26	60.62	58.25

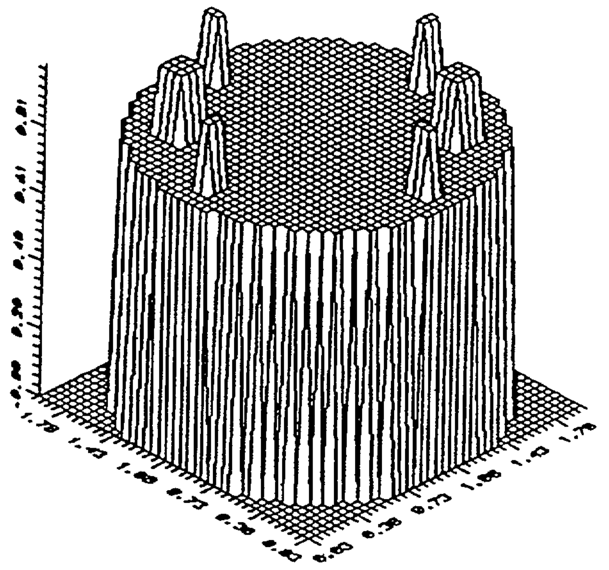
3.3 Sodium Technology

A washing loop for Dumy core subassemblies contaminated by sodium has been built up (photo 2 A and B) in CIAE. The washing process is divided into two steps: fine spray excited by an ultrasonic generator with nitrogen and water rinse. According to our experience, it is a little cheaper to operation and easier to control Na-H_2O reaction if compared with water steam and nitrogen.

A spray sodium fire test has been prepared (Photo 3 A.B and C). To put out sodium fire will be test by the expansion graphite as fire extingushing chemical with presurized nitrogen. The smoke detectors will give the trigger signal. The test will be conducted in the middle of this year.

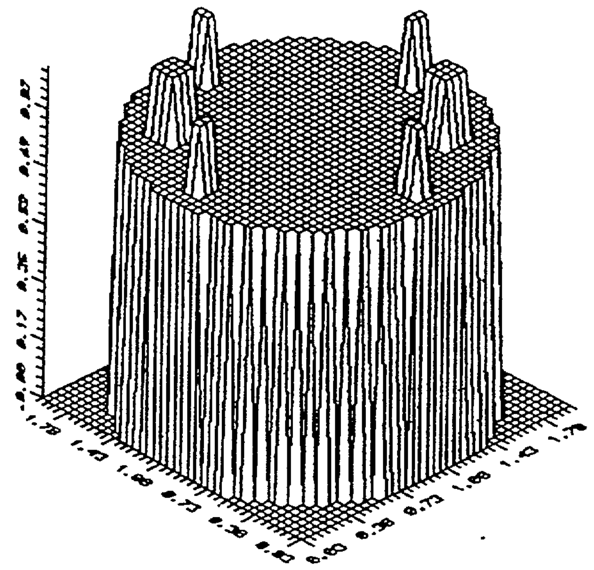
3.4 Materials

316Ti stainless steel has been selected for fuel cladding and core structure material of the CEFR. Even though the trial fabrication of this type of material and some experiments for the aim at the compatibility with sodium have been carried out before. A new fabrication of 316Ti ss is



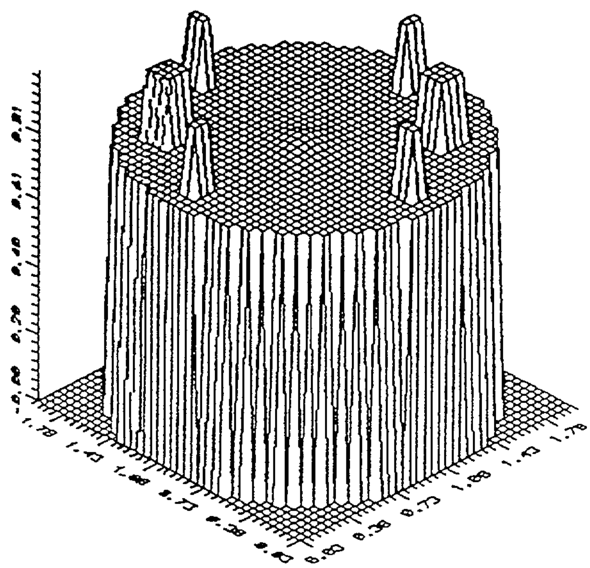
WAVE MAX 0.0000000 m
WAVE MIN 0.0000000 m
WAVE HEIGHT 0.0000000 m

Fig. 1 T=0.0 s



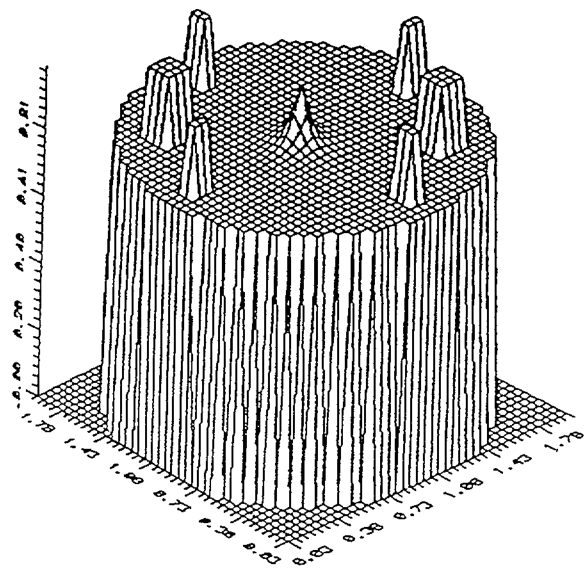
WAVE MAX 0.007132435 m
WAVE MIN -0.006219768 m
WAVE HEIGHT 0.01335220 m

Fig. 2 T=0.1 s



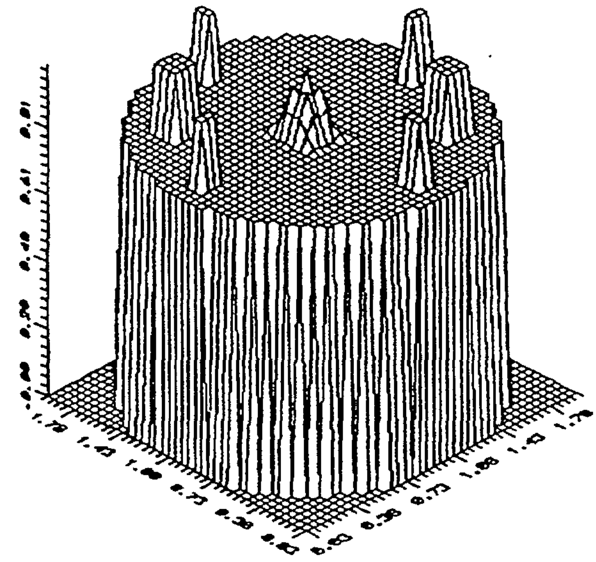
WAVE MAX 0.08686250 m
WAVE MIN -0.01661394 m
WAVE HEIGHT 0.1034764 m

Fig. 3 T=0.4 s



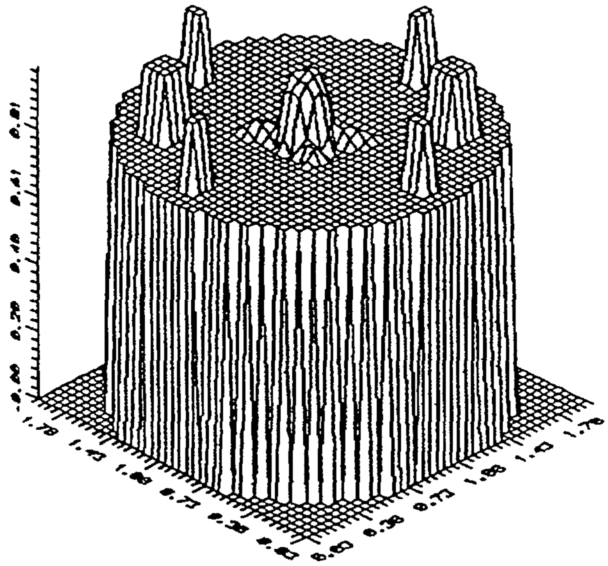
WAVE MAX 0.7644682 m
WAVE MIN -0.02456646 m
WAVE HEIGHT 0.7890347 m

Fig. 4 T=0.5 s



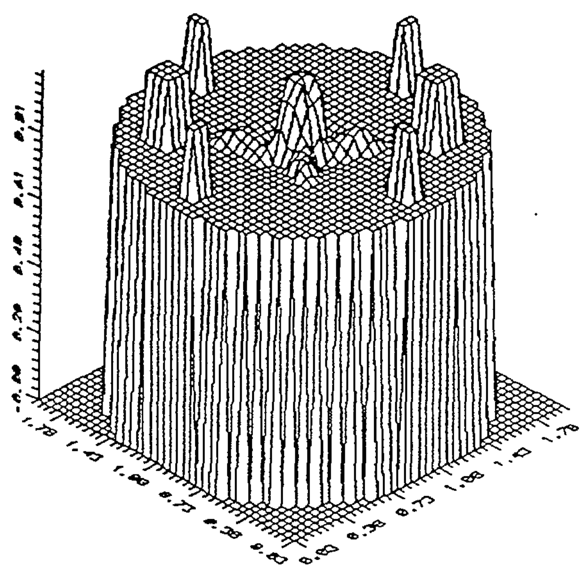
WAVE MAX 0.8799995 m
WAVE MIN -0.02042143 m
WAVE HEIGHT 0.9004209 m

Fig. 5 T=0.6 s



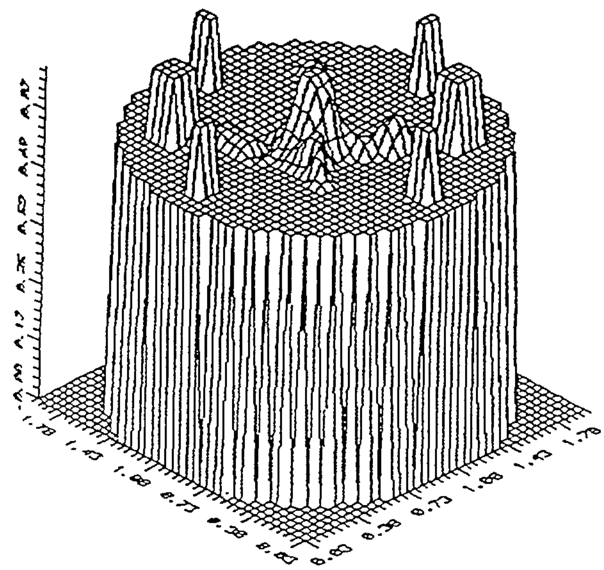
WAVE MAX 0.8799995 m
WAVE MIN -0.1051179 m
WAVE HEIGHT 0.9851174 m

Fig. 6 T=0.7 s



WAVE MAX 0.8799995 m
WAVE MIN -0.1537423 m
WAVE HEIGHT 1.033742 m

Fig. 7 T=0.8 s



WAVE MAX 0.8799995 m
WAVE MIN -0.09832170 m
WAVE HEIGHT 0.9783211 m

Fig. 8 T=0.9s

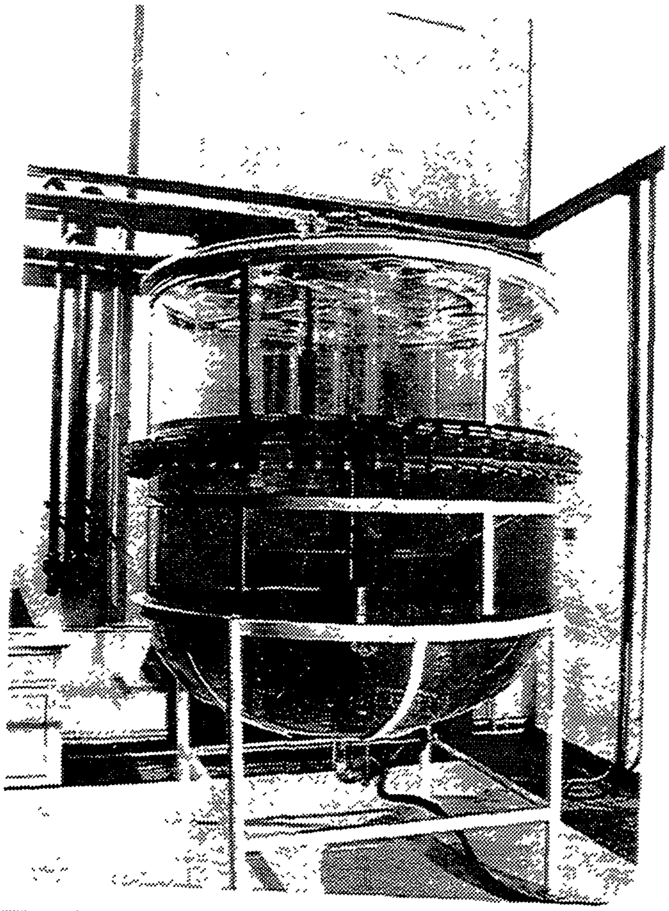
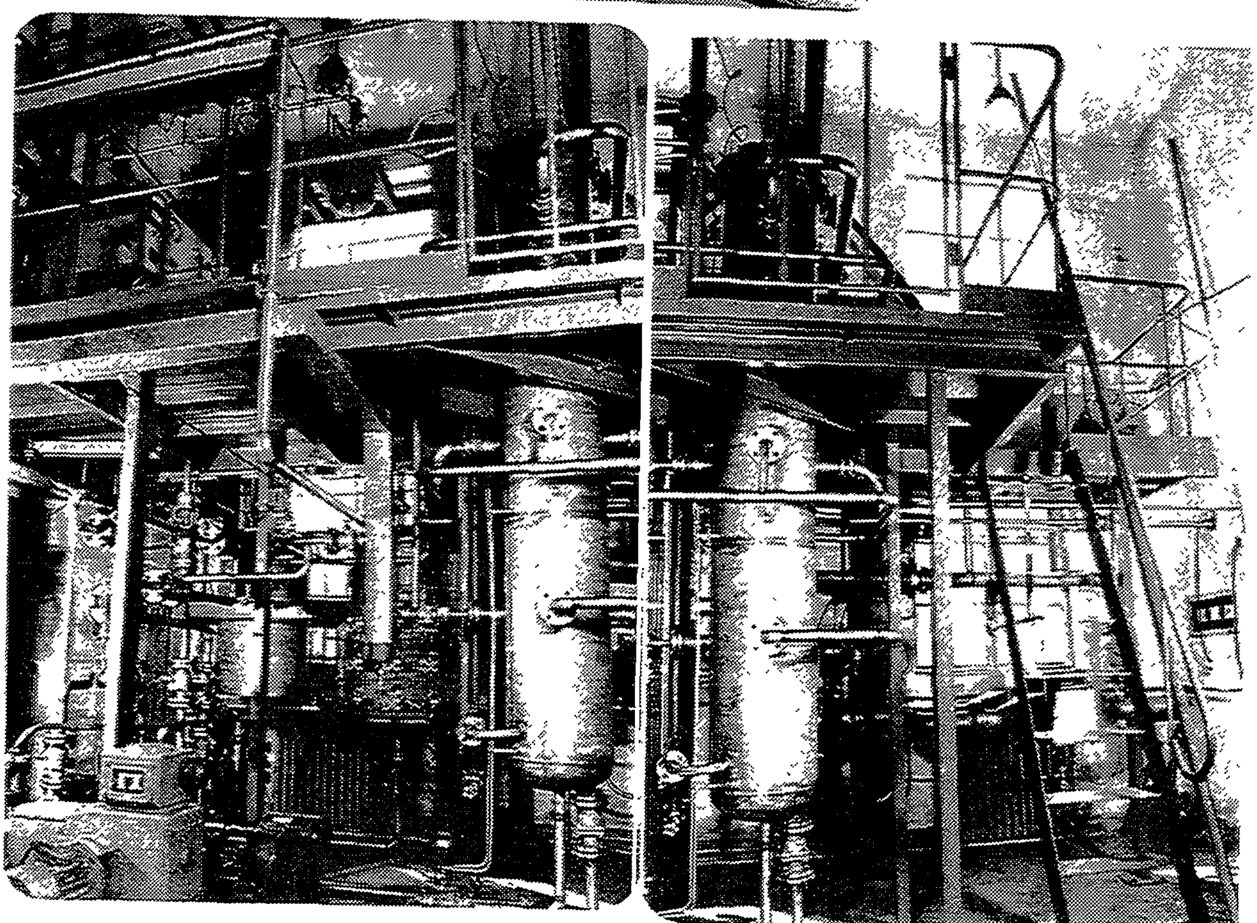
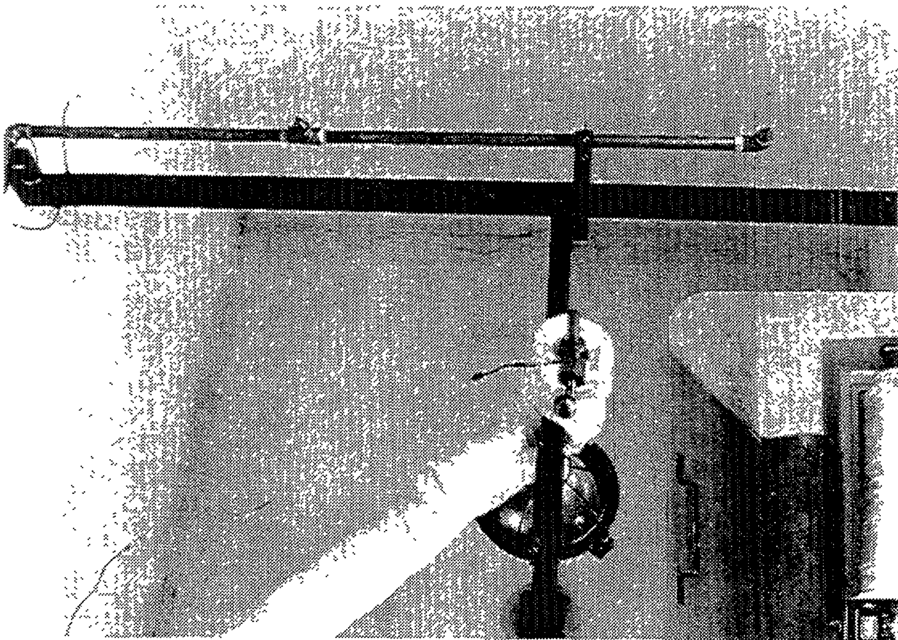


Photo 1
CEFR Water
Mock-up
Diameter: 1.6 m

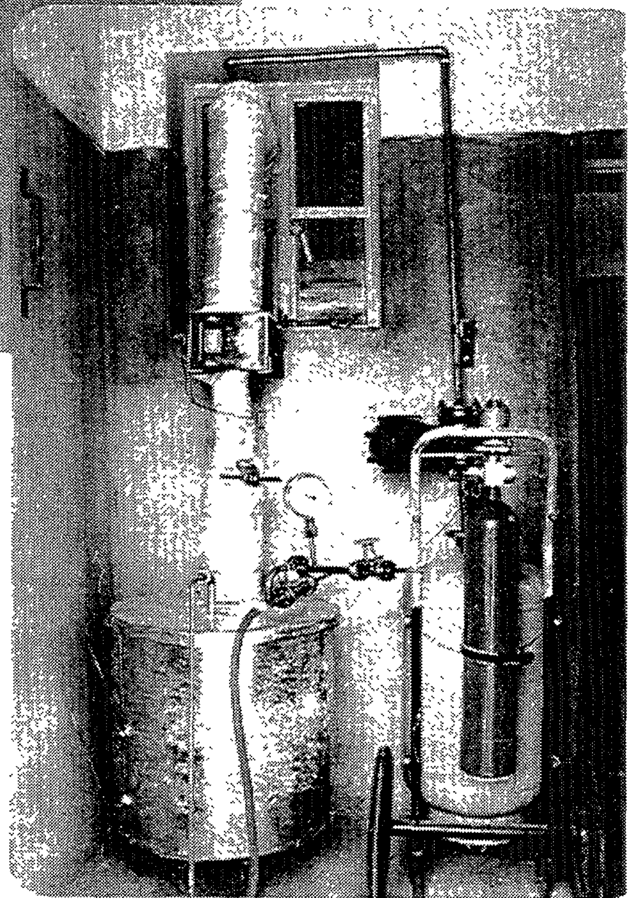


A B

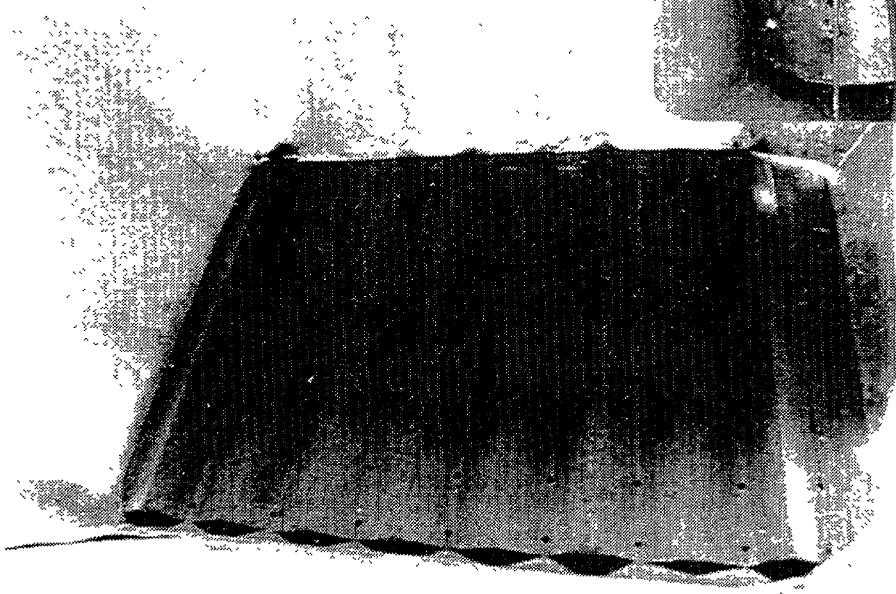
Photo 2 Washing Loop



A



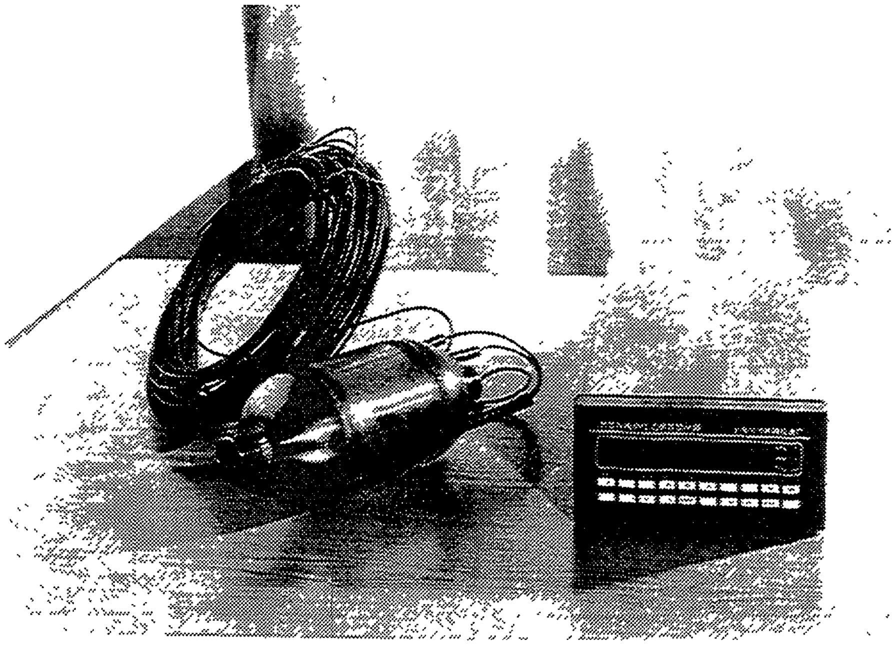
B



C

Photo 3 Na fire Test

Photo 4
Immersed Na
Flow Meter
(Intelligent)



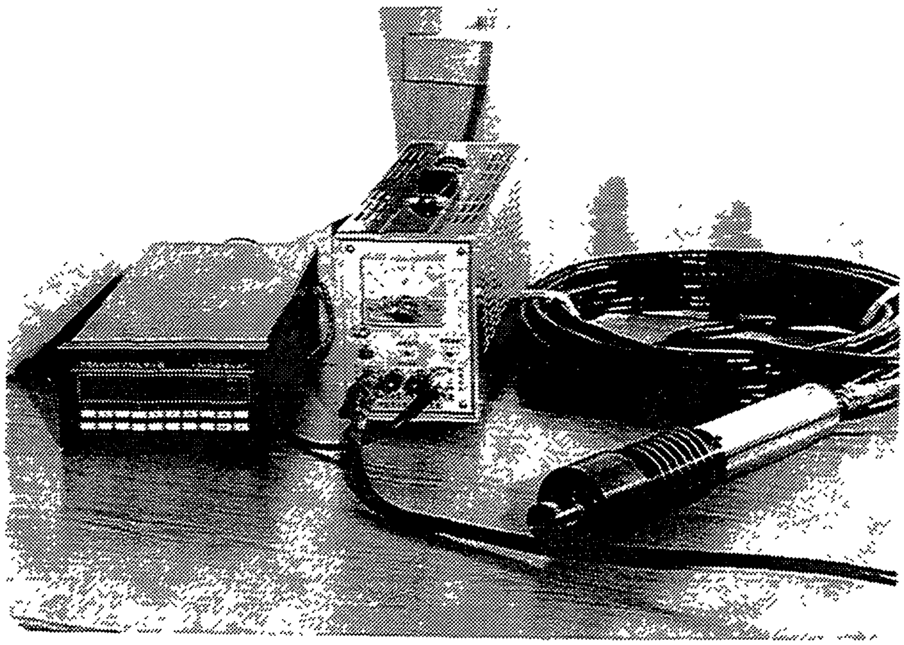
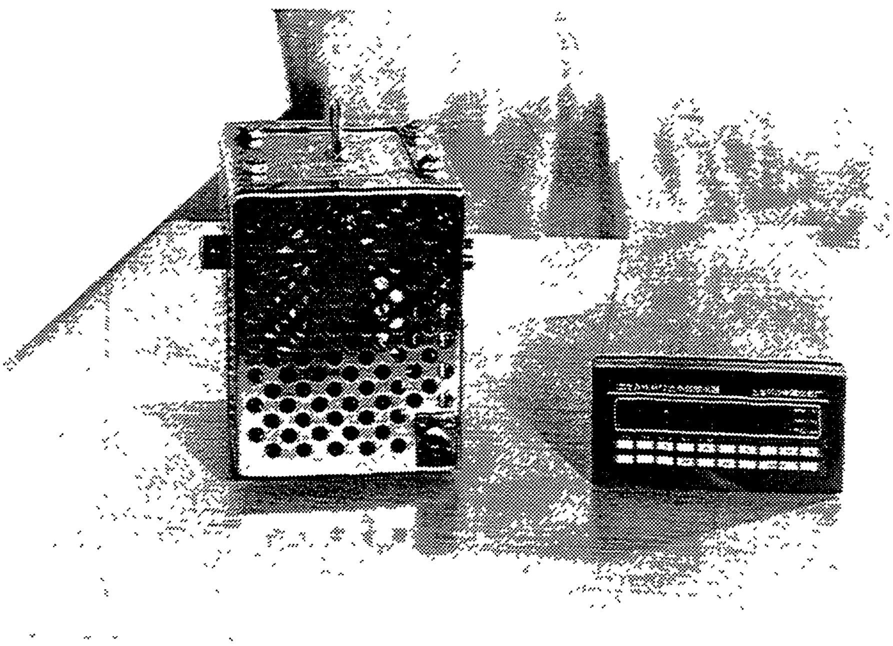


Photo 5
Na Pressure
Transducer
(Intelligent)

Photo 6
Na Flow Meter
(Intelligent)



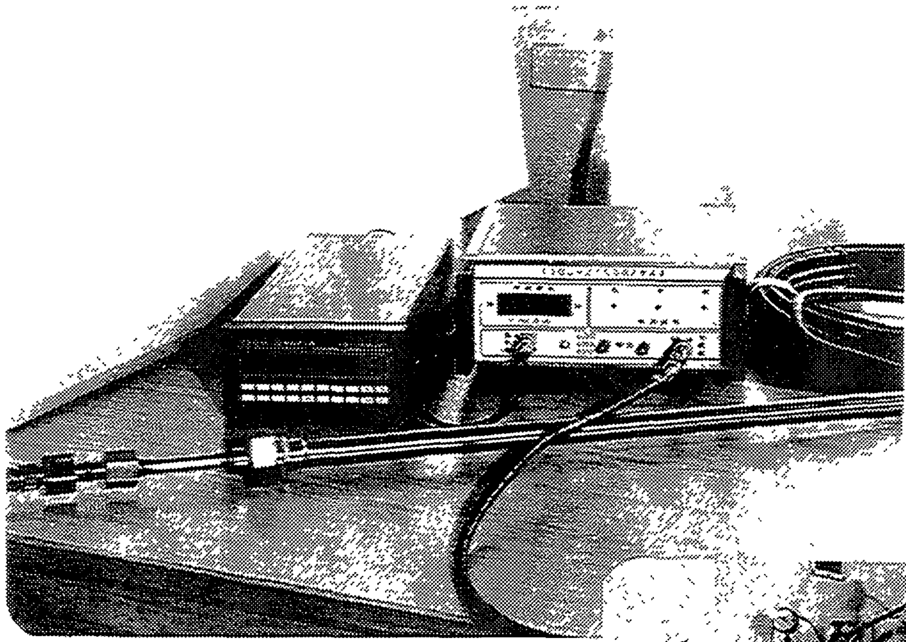


Photo 7A
Na Level Detector
(Intelligent)

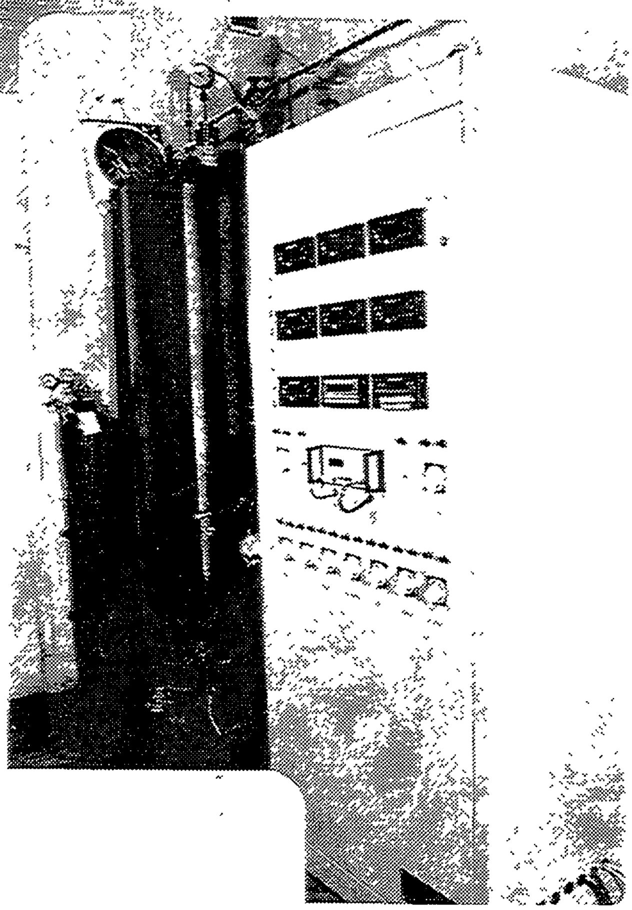


Photo 7B
Na Level Detector
Checking Facility

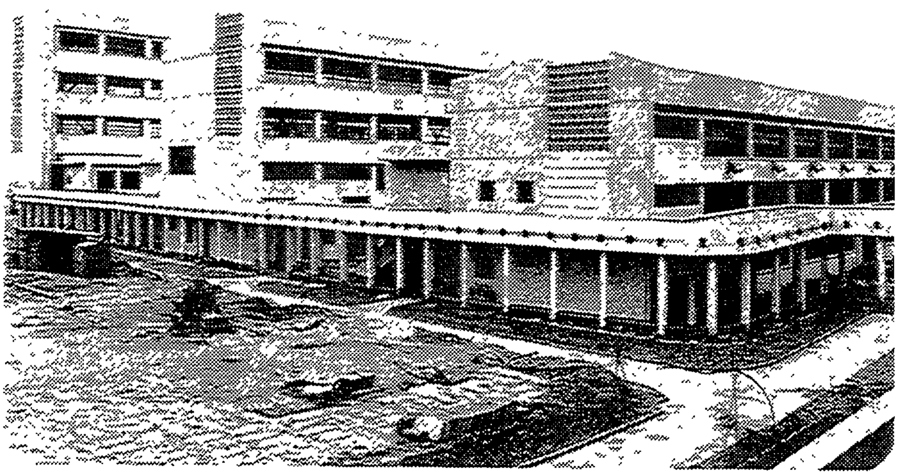


Photo 8
Engineering
Development
Laboratory

started, some sodium loops and facilities as following have been ready to accept the material test specimen:

material corrosion sodium loop;
mass transfer sodium loop;
material creep and fatique sodium loop;
bi-axial creep facility
fission products-cladding interaction facility
modified miniature specimen test (MMST) facility etc.

Concerning the MMST the fundamental relation are established via analysis of elastic-plastic bulge deformation behavior of a middle thick circular plate loaded at the center for which the circumference is fixed and points at the boundary can move in radial direction. The formulas for calculating material strengthes and ductility could be derived from the relation.

Three types of unirradiated materials and a kind of 316 ss irradiated by protons were tested with the MMST.

3.5 Fuels

$(Pu, U)O_2$ in which the percentage of PuO_2 will be 27.2% has been selected for fuel of the CEFR. According to the conceptual design, the core needs 121.6 kg Pu (including 97.2 kg Pu-239) and 93.6 kg U-235 (with 30% enrichment). The pellets of $(Pu, U)O_2$ has been trial-fabricated in the laboratory, which unirradiated performances were as expected.

Concerning the alloy fuel, a small tube type smelting inject casting facility with high frequeucy induction heating has been established about 3 years ago. U-10% Zr alloy of 0.2 kg could be smelted as a batch, with nominal power of 8 kW for the furnace. The maximum temperature of 1973 K could be reached using this furnace. The U-10%Zr alloy fuel pins has been fabricated under following optimized parameters for smelting and inject casting:

inject casting temperature	1543 K
presure (with 2.0 s period)	0.16 MPa
mode lift time	10 s

It was shown by the analysis that the composition of the fuel pin was as expected and following properties have been obtained:

Zr content of 10 ± 0.02wt%;
homogenous phase infrastructure;
even-distribution of two phases αU and UZr_2;
αU grain size of 30 μm and
UZr_2 grain size of 6 μm.

The investigation on the phase transition in U-Zr alloy has been conducted. According to the characteristics of the transformation in the U-Zr alloy, the transition temperature were determined by the Differential Scanning Calorimeter method (DSC) analysis as following: the peritectoid reaction $UZr_2 \longleftrightarrow \alpha U + \gamma_2$, the eutectoid reaction $\beta U \longleftrightarrow \alpha U + \gamma_2$ and the monotectoid transition $\beta U + \gamma_2 \longleftrightarrow \gamma_1 + \gamma_2$ are occurring at 887 K, 934 K and 964 K respectively. The temperature of the boundary between the single phase field and the miscibility $\gamma_1 + \gamma_2$field are 983 K, 990 K and 992 K respectively corresponding to the U-Zr alloy containing 7.99, 9.31and 10.80 weight percent zirconium. The enthalpy changes of these phase transitions in U-10 Wt%Zr alloy are about 500 kJ/mol 90 kJ/mol 4050kJ/mol and 250 kJ/mol.

3.6 Sodium Instrumentations

For measuring the sodium flow rate of primary loop in the CEFR, it is intended to equipe an immersed sodium flow meter to the straight pipe left from primary pump. As its first prototype, a small immersed sodium flow meter with 3 m^3/h full scale has been developed (photo 4)in CIAE, Its basic error is less than ±2.3% of full scale.

An intelligent sodium presure transducer of 1 MPa full scale based on a ring measuring force has also been developed (photo 5) in CIAE. Its linearity is satisfactory and the basic error less than ±0.3% of full scale.

Recently an intelligent sodium flowmeter (photo 6) and sodium level meter (photo 7A) and its checking facility (photo 7B) have been developed in CIAE. The intelligent means the meters with on-line compensation for environmental temperature and with alarm function when it reached the preset value.

3.7 Engineering Development Laboratory

Since the end of the year 1990 it was started to build the Engineering Development Laboratiry of 18000 m^2 construction surface. The EDL includes three main buildings which will be used for sodium technology development, thermohydraulics and safety research and development and demonstration and proving of facilities and components respectively. (photo 8)

4. Conclusion

In the year 1994, the CEFR project was still waiting for its approval by the State Planning Commission. Its budget trouble has happened. But it is good that recently the conclusions of a series meeting about the suitability to design and construct now the CEFR are positive, it will promote the CEFR's approval by the SPC.

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FAST REACTOR DEVELOPMENT PROGRAMME IN FRANCE DURING 1994*

C.L. RIGOLEUR
Commissariat à l'Energie Atomique,
Centre d'Etudes de Cadarache
France

Abstract

Activities on LMFR technology have been conducted in France in close cooperation with Western European Countries. Status of technology development in 1994 and future R&D are described in this paper. The paper also presents some performance results and gives an overview of the European Fast Reactor, the CAPRA project and knowledge acquisition programmes as well as the Phenix and Superphenix reactors restart and operation experience.

1 - GENERAL SITUATION

In 1994, the total electrical production in France was 454 TWh, out of which 341.8 TWh (75 %) were produced by nuclear power plants, 32.2 TWh (7 %) were produced by conventional thermal plants, 80 TWh (18 %) were produced by hydraulic plants.

The exported power was 63.4 TWh, the energy used in pumping was 3.1 TWh and the line losses were 29 TWh. Consequently, the net electrical power consumption in France was 358.5 TWh corresponding to a slight increase of 0.8 % in comparison to the 1993 consumption.

The availability factor for all 900 and 1300 MWe PWR improved slightly up to 81.3 % (in comparison to 80.6 % in 1994) mainly due to better management of shutdown periods (12.1 % instead of 14.9 % in 1993).

At the end of 1994, after the closing in May of its last gas-cooled reactor at BUGEY, "Electricité de France" had 54 PWR units totalling 57150 MWe.

Control of the defects detected in the penetrations of the vessel heads was almost complete and 6 vessel heads are replaced each year without any consequences on the availability of the plants.

In the field of Fast Reactors, the main events of the 1994 year were the following.

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* *With contributions from Messrs. B. Mesnage (CEA), A. Roux (NERSA) for the Super-Phenix paragraph,*
P. Anzieu (CEA), R. Del Becarro (EdF) for the "Knowledge acquisition programme",
J. Rouault (CEA for the CAPRA paragraph,
J.C. Lefevre (EFR Associates), C.H. Mitchell (EFR Associates), G. Hubert (EFRUG) for the EFR paragrpah.

At the end of December the PHENIX reactor was authorized to perform the 49 cycle at 350 MW th. This cycle was reached at the beginning of April 1995 (see § 2).

Concerning the CREYS-MALVILLE plant (SUPER-PHENIX) the work to improve resistance to large sodium fires and sodium leak detection on secondary circuits was accomplished by mid - 94. The NERSA society was authorized to restart SUPER-PHENIX. This was formelly carried out by a new "Décret d'Autorisation de Création" (a new basic nuclear licence) issued by the French Government on July 11, 1994 and by a routine regulatory permission to operate granted by the ministers of industry and environment, advised by nuclear safety agency after a final safety review. Finally the criticality was reached on August 5, 1994 and the reactor power was increased in successive steps (see § 3).

The "Décret d'Autorisation de Création" stipulates that because of its prototype character, SUPER PHENIX will have to be operated in conditions explicitely giving priority to safety and knowledge acquisition, with an objective of research and demonstration.

In this context the so-called "knowledge acquisition" programme (§ 4), proposed by NERSA, "Electricité de France" and "Commissariat à l'Energie Atomique" is designed to prove the capacity of a large FBR to produce electricity on an industrial scale, to test the consumption of plutonium and minor actinides in a large fast reactor, as well as to provide information on technology of sodium-cooled fast reactors.

The CAPRA programme, initiated on February 1993, aimed at demonstrating the feasability of a fast reactor to burn plutonium as high as possible. The first two-year phase of the CAPRA project studies (1993-1994) was achieved and will be presented in § 5.

The European Fast Reactor was launched in 1988 and has reached an important stage with the completion of the Concept Validation Phase. The status of the project will be presented in § 6 as well as its evolution.

The four main objectives can be summarized as follows :

- A good operation of PHENIX (§ 2) and SUPER PHENIX (§3).

- The "knowledge acquisition" programme (§ 4)

- The CAPRA programme (§ 5)

- The EFR programme (§ 6)

These objectives must be placed in the prospect of a decision to construct a fast reactor which will be made around 2005 (cf B. BARRE presentation at EUROFORUM Paris, January 25-26 th, 1995)

The Rand D in support of these objectives is presented in § 7.

This Rand D must also be placed within the context of international collaboration. Specific agreements exist between Europe and Japan and beetwen France and Russia. (CEA-MINTOM agreement where Germany and the United Kingdom are "Associated Partners").

2 - PHENIX

For the same reasons explained in the "Review of the IWGFR activities for the period from the 26 th Annual Meeting" major decisions were taken early in 1994 namely :

- the order of three extra IHXs ; they will be delivered no later than spring 1997

- the replacement of the 321 stainless steel pipes in the secondary circuits. This work on the secondary loop SL 2 was started in 1994 and will be completed in 1995 . The work on SL 1 will start in 1995 and in1996 on SL3. When the new IHXs will be delivered in 1997 the three loops will be available

In 1994 the expansion tanks of the secondary loops (SL1, SL2, SL3) were repaired

The re-inforcement in the sodium leak detection systems was completed early 1994

The installation of an insulator covered by steel plates on the floor of the gallerie SL1 in July 1994, completed the work of protection against sodium leaks in the gallerie SL1. Similar work is being carried out at present in gallerie SL3.

Each of the sodium loop SL1 and SL3 was equiped with a new IHX and an IHX with a few thousand hours of operation or a few ten thousand hours of operation.

Consequently at the end of August 1994 the secondary loops SL 1 and SL3 were available .

Finally the plant was ready for operation at the end of September 1994

After a technical review by the Safety Authorities the authorization to perform the 49 th cycle at 350 MWth was given on December 21, 1994. This 49 th cycle was accomplished without any problem on April 7, 1995.

As previously stated the sodium loop SL2 will be repaired in summer 95 and will be equiped with two IHXs (new and with a few thousand hours of operation).

Complementary protection against sodium fires will be set-up on the secondary loops in the steam generator building.

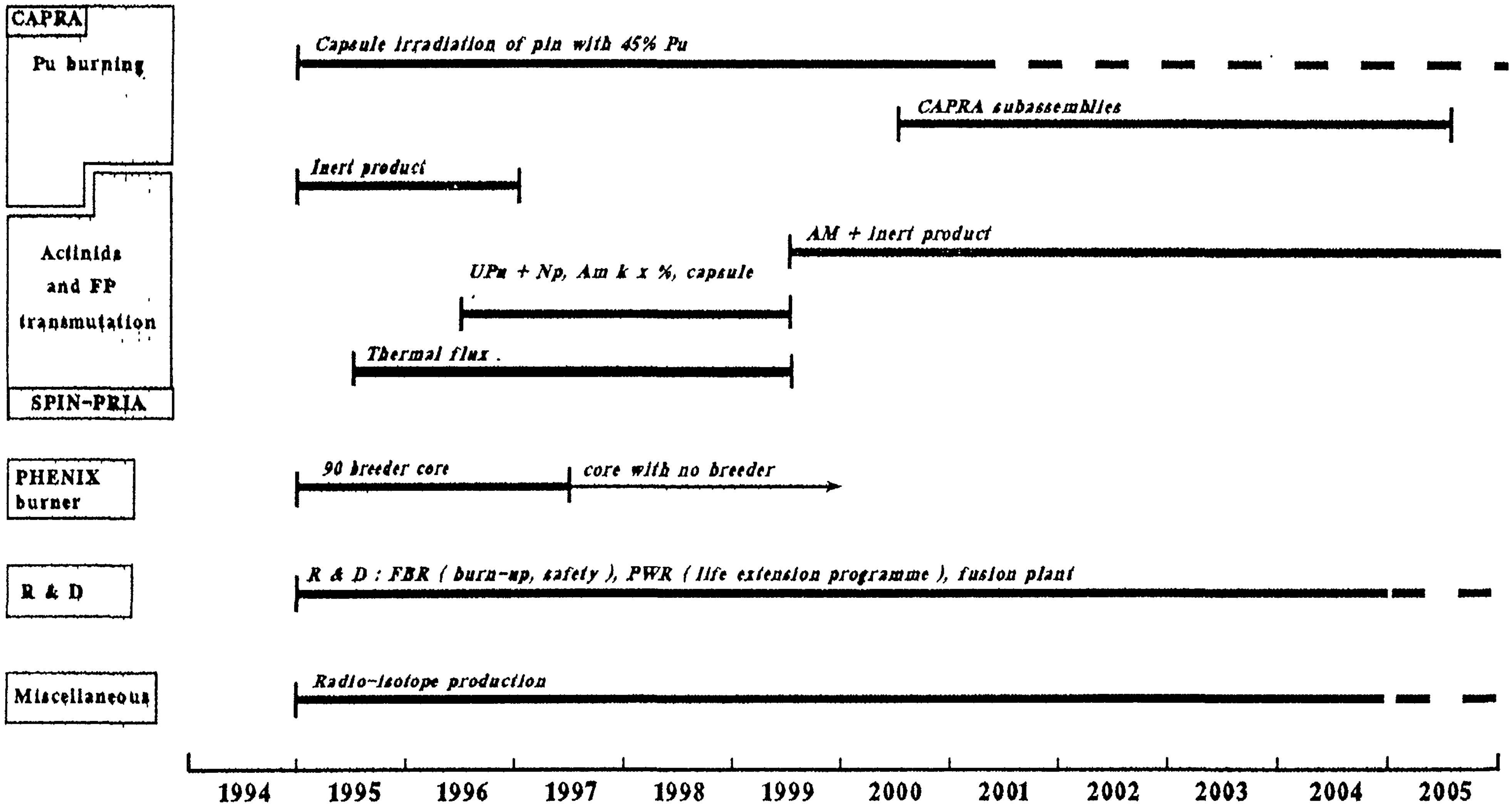


FIG. 1. Next ten-year irradiation programme.

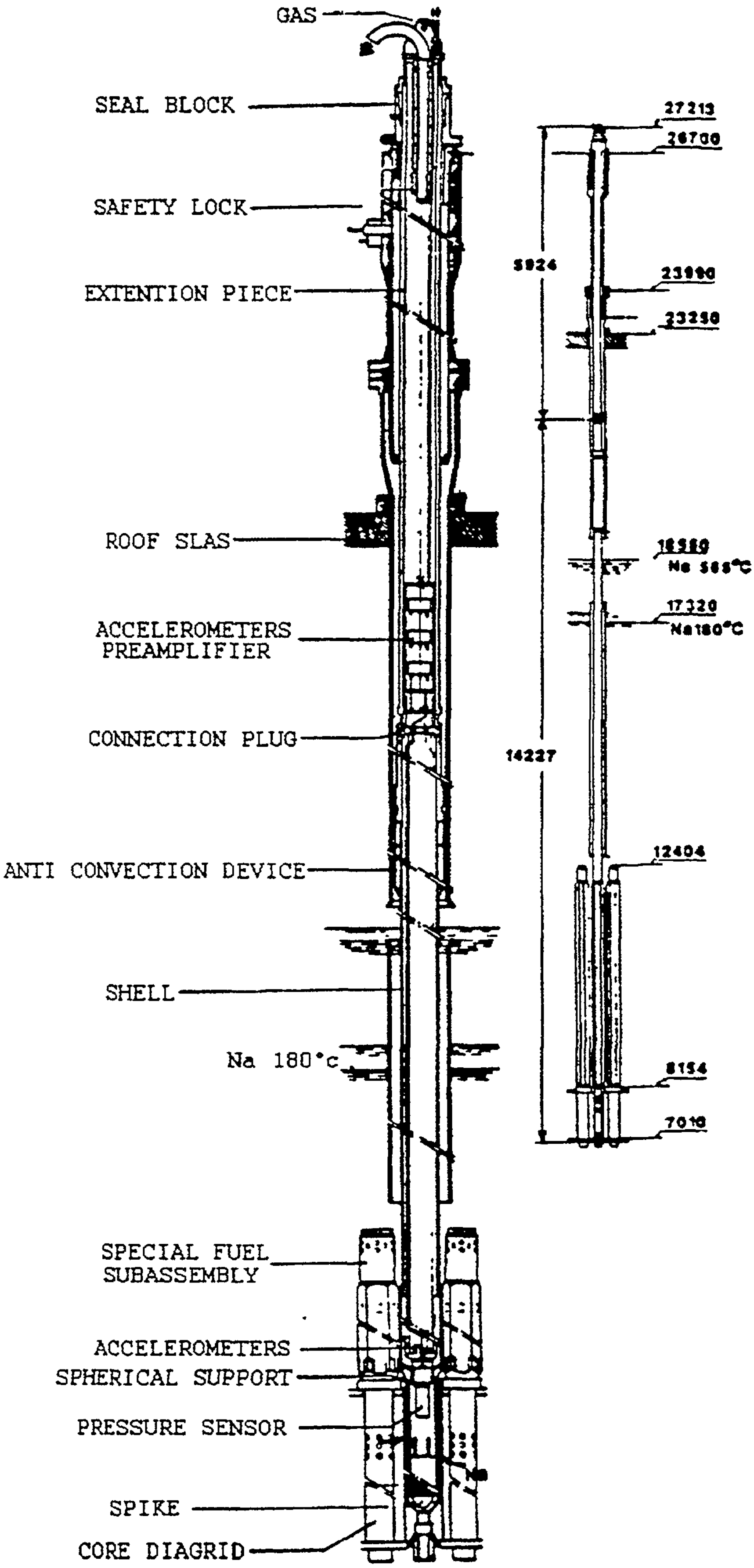


FIG. 2. CREYS-MALVILLE BOUPRESS device.

The plant will be ready to start the 50 th cycle in October 1995 but the authorization must be granted by the Safety Authorities

The year 1994 was also extremely rich in information resulting from the analysis of the defects observed in the expansion tanks (for details see the papers presented at the Technical Committee Meeting in Aix en Provence) and in the IHX with 70 000 hours of operation.

For general information the next ten year irradiation programme in shown in fig 1. and is discussed partly in the CAPRA programme paragraph.

PHENIX life extension project

Extending the life time of the reactor for another 10 years is planned for three purposes :

- to qualify high burn-up oxide fuel

- to perform the necessary experimental irradiation in support of the CAPRA project (enhanced plutonium consumption) and SPIN (separation and incineration of long-lived wastes) fig. 1),

- to supplement the operating experience feedback and in particular to better understand the negative reactivity incidents and the ageing process of the structures better.

In order to complete this programme the reactors needs to operate 55 000 hours more. After completion of this 55 000 hours the reactor will have 152 000 hours of operation and this is in excess of the 140 000 hours value in the Safety Report (precisely 110 000 hours at nominal power, 15 000 hours with 4 IHXs out of 6, 15 000 hours with 2 primary pumps out of 3).

It is necessary to justify this increase of operating time considering :

- the reinforcement of safety requirements

- the enhancement of knowledge

- the feedback experience (for example in secondary circuits)

- the characteristics of the future operating conditions.

Consequently the "life extension project" will include :

- an updated evaluation of the state of the reactor, incorporating the accumulated experience since its first criticality,

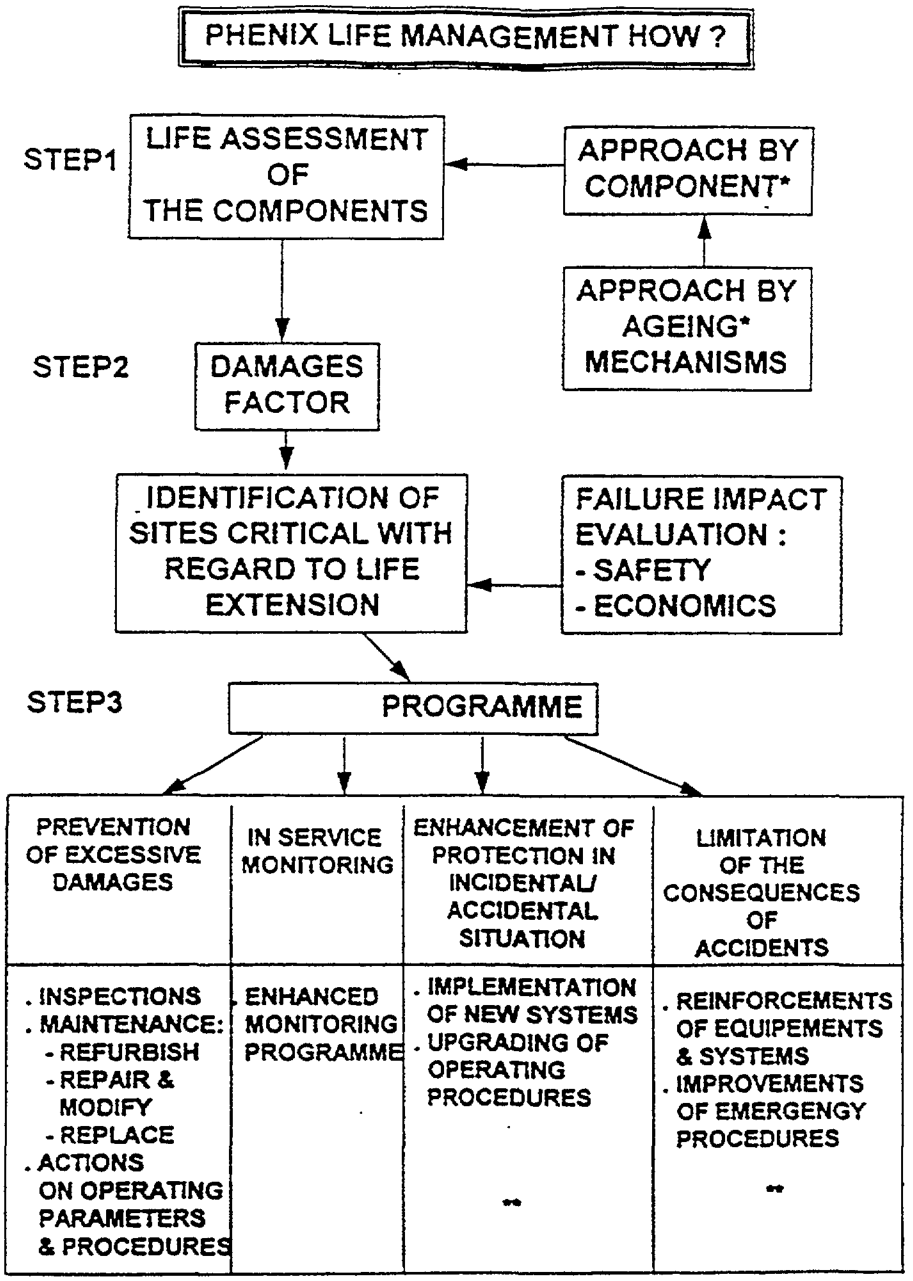
- the definition of a working method to comply with requirements of the Safety Authority, based on similar studies performed by EdF for PWRs,

- a precise planning of the necessary tasks to be performed by CEA and by the industrial partners,

- a periodic synthetic evaluation of the results in order to constitute the necessary safety documents and to identify the necessary actions on the systems, components and operating conditions,

- an evaluation of the operational feedback, in particular in view of future Fast Reactor concepts.

A general presentation of the necessary steps is shown in the following table.



* : including feed back experience ** : closely related to the evolution of safety requirements

3 - CREYS-MALVILLE POWER PLANT YEAR 1994

For Superphenix, 1994 was the year for resumption of power, which took place in August after four years of upkeep and safety improvement works and administrative procedures.

On 22nd February 1994, the Government in fact took position in favour of resumption of the Plant by realigning and licensing once again its mission as an industrial prototype: a reactor devoted to research and demonstration, and operated within the context of a know-how feedback programme based on two priority objectives: research on the use of plutonium and reduction of long-lived radioactive waste. The reactor , on the other hand, should evolve as rapidly as possible towards subgeneration as consumer so as to limit the quantities of plutonium produced.

This decision was based on the CURIEN Report of December 1992, specifying the conditions under which Superphenix could contribute to incineration problems of nuclear waste, and on the Report of the Safety Authorities, delivered on 18th January 1994, declaring in particular that the safety level of Creys-Malville was the same as the 54 PWR reactors of the French nuclear park.

3.1.PREPARATION FOR RESUMPTION OF POWER

The first half of 1994 was therefore given over to the preparation of resumption of power with, first of all, completion of improvement work for prevention of significant sodium fire in the secondary galleries and steam generator buildings which were begun at the end of 1992; and various works of maintenance and modification, in particular on the evacuation line of spent subassemblies, started at the same time.

The "sodium fire" works, were undertaken with the secondary loop drained, and their completion during the months of February to May saw the sodium filling of the four secondary loops respectively. This allowed, during the second fortnight of May, carrying out the isothermal test campaign indispensable after the long shutdown of the installation and the modifications in the secondary galleries.

These isothermal tests at 350°C allowed checking the behaviour of the secondary circuits and the new "sandwich type" sodium leak detectors during the rise in temperature, and to control the hydraulics of the primary circuit by taking pressure and vibration measurements in the diagrid using the special BOUPRESS device (figure 2), which was also used for startup tests of the reactor in 1986. Furthermore, at the time of installation of this device, which takes the place of a fuel subassembly, a sound operation test of the primary handling line was carried out after upkeep works.

Sodium refill of the loops also allowed handling components on the reactor bloc necessary for completion of the reinstatement programme of primary plugging indicators and clad rupture detection modules.

All this period was also beneficial :

- on the one hand, for carrying out several statutory tests or checks on the numerous components of the installations. In particular, the leaktightness test on the third boundary was realized : in January, the safety tank under azote pressure of 100 mb, then in June, the dome under air pressure of 100 mb.

- and on the other hand, for the preparation of a startup programme, to be run, at the request of the Safety Authorities, in a very progressive manner, with a view to validating the improvement measures concerning operating conditions, proposed following the incident of sodium pollution in June 1990. At the same time a specific organization grouping together NERSA, EDF, NOVATOME and the Atomic Energy Commission (CEA) was set up for the follow-up of the installation's behaviour and the detailed know-how feedback of evolution criteria.

This preparation phase for startup was completed in June with the progressive starting up of the feed water plant, seismic updating of the reactor containment revolving crane required by the Safety Authorities and by further interventions on three secondary loops which had to be drained again, then filled before startup.

These last interventions corresponded to two technical problems encountered during verifications on the installation, and which had only partially been carried out before the isotherm tests. These concerned ;

- eliminating superficial defects detected on some connections of the sodium storage tanks of the secondary loops, by grinding,

- replacing the argon pressure balancing pipes of the space between the rupture disks of the steam generators as one showed cracking on a weld (figure 3).

3.2. REACTOR STARTUP

The signing, on 11th July by the Prime Minister of a new Approval Order gave the Plant legal existence again without which no authorization to operate could be given.

The Order confirmed the new orientations given by the Government and elaborated in the know-how feedback programme committed by NERSA, EDF and CEA and submitted to the approval of two scientific experts nominated by the Government.

This programme, which represents the basic feature of the future operation at CREYS-MALVILLE, has three further objectives :

- demonstrate the capacity of a fast reactor to produce electricity on an industrial scale, operation of the prototype enabling the Operator to follow various operating behaviour for fuel, systems and components, and in-service inspection,

- assess the workings of this type of reactor as a net consumer of plutonium according to two demonstration modes: progressive core conversion from the fast breeder mode to the burner mode, and qualification on complete subassemblies of technical solutions developed in the CAPRA project,

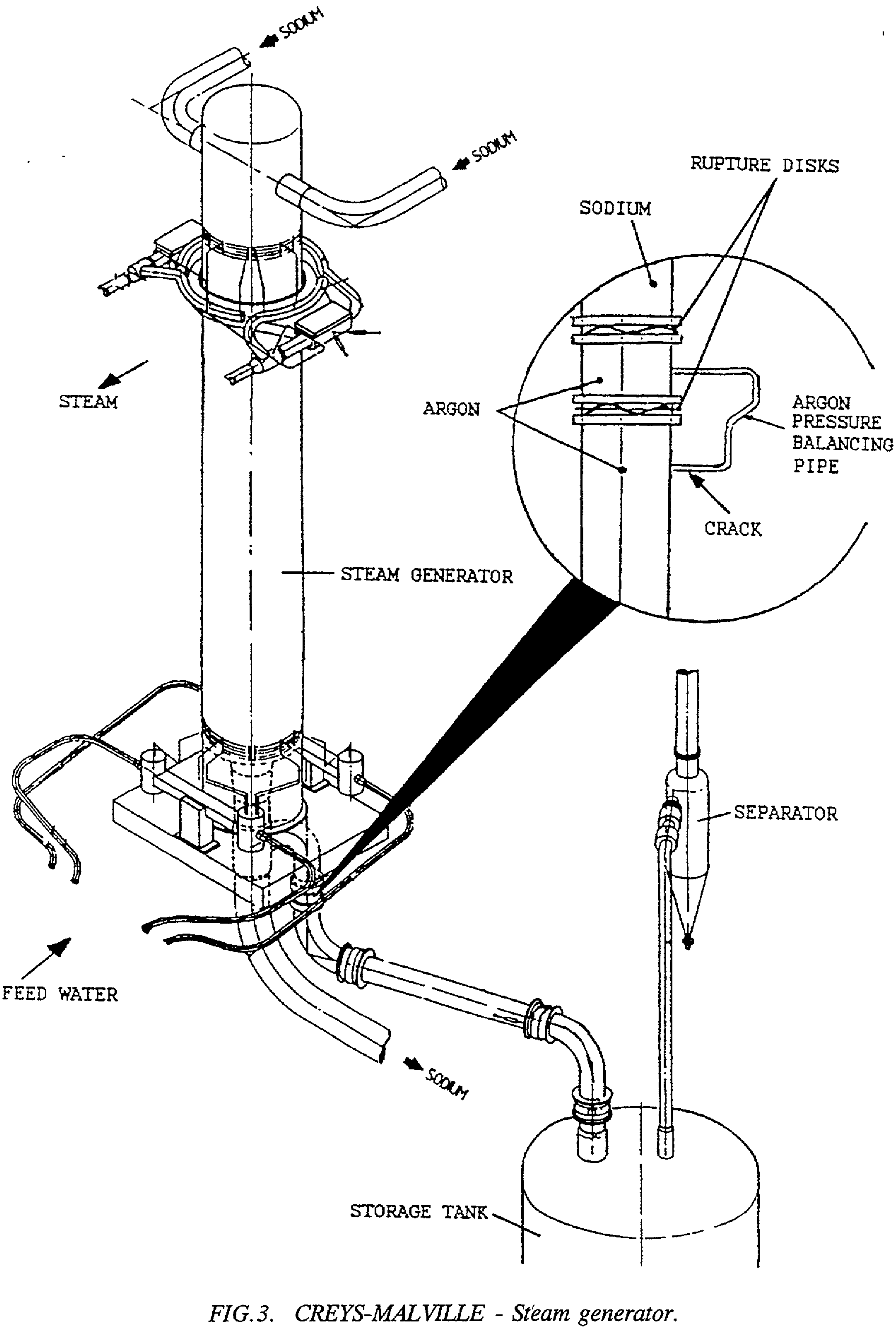


FIG.3. CREYS-MALVILLE - Steam generator.

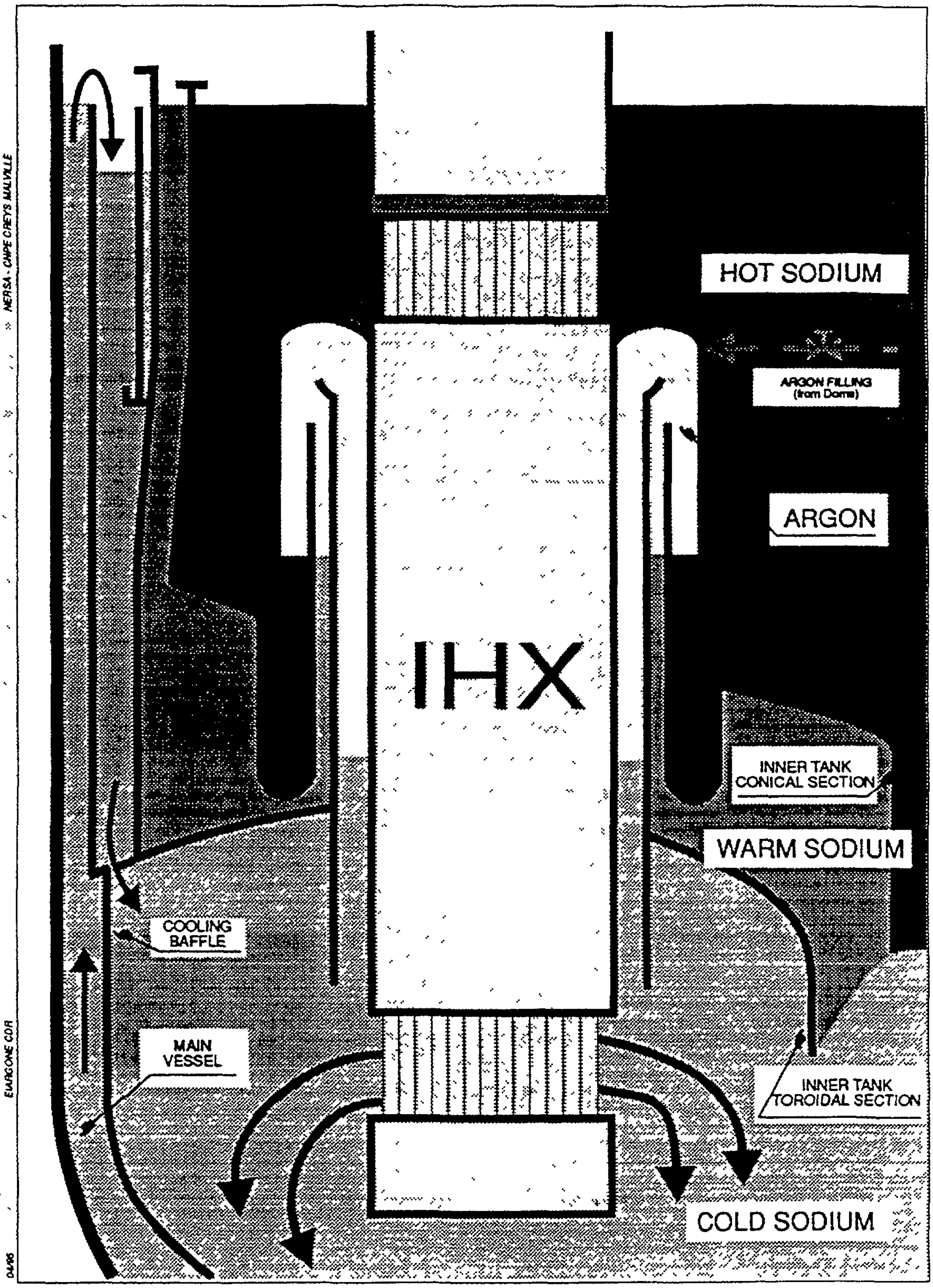


FIG. 4. CREYS-MALVILLE - IHX argon bell.

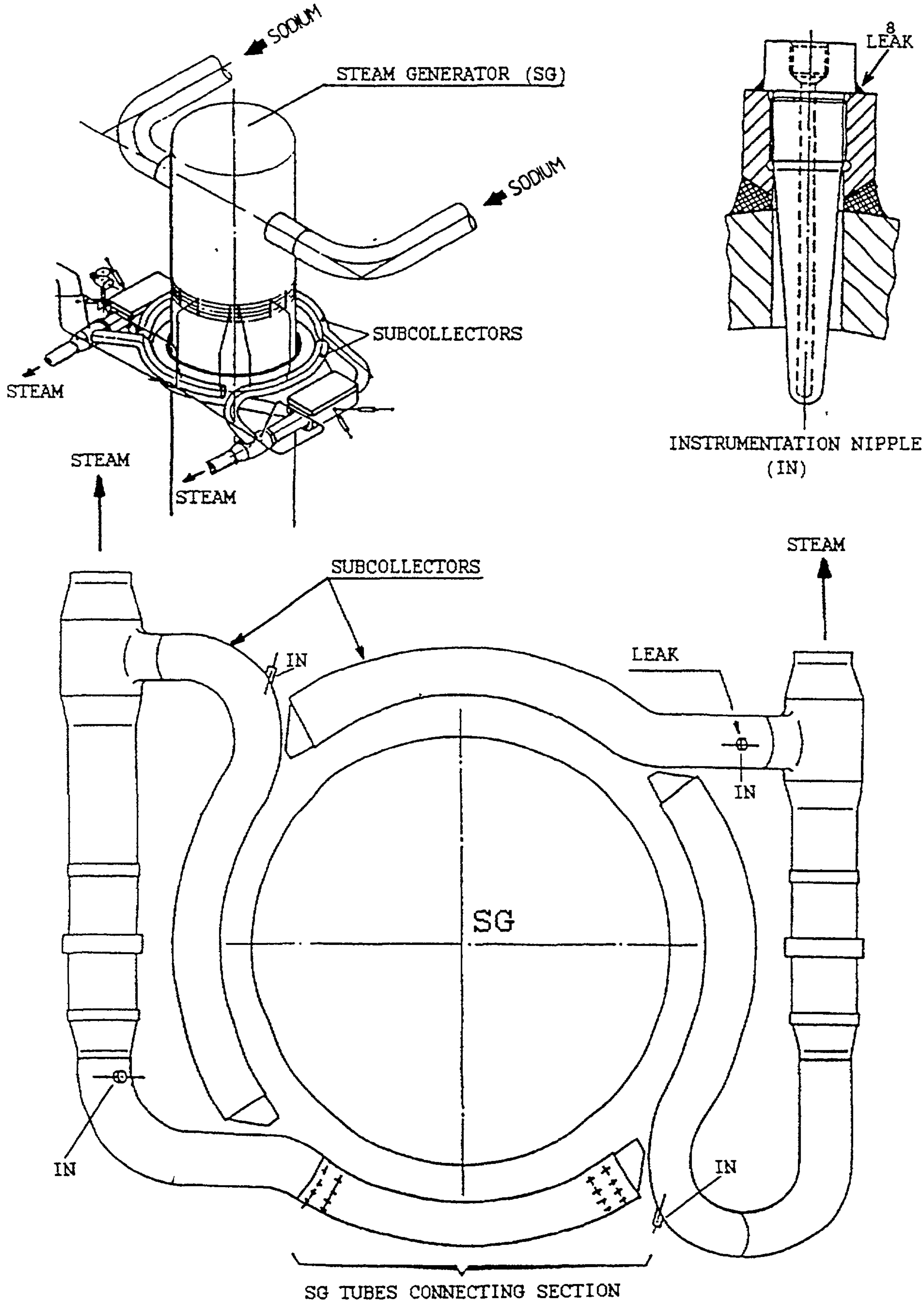


FIG.5. CREYS-MALVILLE - SG steam collector.

- study the possibilities of long-lived radioactive waste destruction with a triple objective: confirm the capacity of the fast reactor to incinerate, confirm the performance of this solution and show the compatibility of the two methods of approach (plutonium burning and actinide incineration).

In these conditions, and after favourable opinion of the Safety Authorities, the Ministers concerned granted, on 3rd August 1994, approval to restart the reactor, subject to a progressive programme step by step, proposed by the Operator, but bound nevertheless by several compulsory successive approvals to be given by the Safety Authorities, to build up from 3% to 30%, then 60% and 90% of nominal power.

Reactor criticality took place on 4th August and was followed by a neutronic test phase at less than 3% power, carried through at various temperature levels between 180° C and 345° C. These tests showed that the behaviour of the Superphenix core remained conform to forecasts, and that the main and auxiliary sodium circuits operated in a satisfactory manner. The 345° C temperature level was put to advantage to check the calibration by hydrogen injection of the steam generator leak detection systems.

This short test phase was prolonged into September and October by an investigation campaign on a phenomenon of abnormal pressure drop in the argon bell of one of the 8 intermediate heat exchangers (figure 4), a phenomenon which had appeared in May but which became less important at 180° C. These investigations led to confirmation of a very slight argon leak in sodium without in fact localizing it accurately, the leak disappearing again.

After analysis, which showed that this new abnormal state of affairs had no significant impact on reactor safety, the Safety Authorities gave their approval on 7th November 1994 to increase reactor output to 30% of its nominal power. This second phase in the resumption of power was mainly devoted to follow-up of sodium physicochemical evolution during temperature build-up, and to the progressive running of the feed water plant.

This second phase of startup rapidly confirmed favourable evolution of sodium characteristics as expected, but it was interrupted on 16th November after steam generator water-fill by the appearance of a very small steam leak on a tightness weld of an instrumentation nipple, located on one of the steam subcollectors at steam generator outlet on C loop; this incident meant once again draining the 4 secondary loops to carry out weld checks on the other steam generators and repair of the faults observed (figure 5).

Startup on 7th December was then disrupted by the brutal reappearance of the argon leak; following a spurious shutdown, caused by a fault in condenser vacuum, reappearance which finally led, considering its evolution in the course of power buildup to 15% of nominal power and connection to the grid of turboalternator train A on 22nd December, to the decision to shut down the reactor on 25th December to undertake necessary actions to eliminate this argon leak, either by intervening directly on the intermediate heat exchanger, or by replacing the latter by the spare heat exchanger present on-site.

Lastly, at the same time as these operating activities, various adjustments not only on the new subassembly handling line but also on the unloading equipments and storage of spent subassemblies continued for preparing the future handling campaigns and particularly the replacement of fertile subassemblies planned for 1996.

4 - KNOWLEDGE ACQUISITION PROGRAMME

4.1. - Objectives

The knowledge acquisition programme, necessitating the operation of the Creys-Malville plant, has three major complementary objectives :

- to demonstrate the capacity of a fast reactor to produce electricity on an industrial scale while contributing to plutonium management and the reduction of long-lived radioactive waste ;

- to study the flexibility of fast reactor using plutonium as fuel and to qualify the technical solutions developed within the framework of a research programme whose major concern is to operate this type of reactor as a plutonium consumer.The CAPRA project in specifically involved in this case (see § 5) ;

- to study the possibilities of destroying long-lived radioactive waste and in particular the minor actinides, americium and neptunium within the framework of the SPIN programme.

4.2. - Research Programme

This programme hinges on three major research objectives defined above.

4.2.1. - Demonstration of a fast reactor prototype in operation

The reactor is operated as a prototype. All observations whether they concern normal or abnormal operating conditions of the reactor are used to make an analysis, to reach conclusions and findings allowing feedback. The specific goal is to prepare the definition of an improved, future generation of fast reactors.

Follow-up on the state of the performances of the different systems and components, in the NSSS involves in-situ measurements, periodic inspections, and special examinations during maintenance along with specific test periods.

This technological knowledge acquisition concern the major components of the NSSS.

4.2.1.1. - Fuel

Obtaining a high burn-up is an imperative for the economy of fast reactors and this objective will be pursued on Super-Phenix in order to optimize reactor operation and to illustrate the actual possibilities of the reactor type. It implies thorough knowledge acquisition of the behaviour of the materials in the core sub-assemblies.

4.2.1.2. - Sodium circuits and reactor block

The experience acquired in the operation of sodium circuits having dimensions similar to those found on the Super-Phenix reactor constitutes a fundamental source of information concerning several points :

- Testing the quality of the sodium.
- Contamination transfer within the circuits.
- The decontamination and washing of the unloaded components.
- Guarantee of the system's capacity to detect possible sodium leaks on the pipes.
- Thermo-mechanical and thermo-hydraulic behaviour of the sodium-coolant pipes.

4.2.1.3. - Steam Generators

The steam generators used at Creys-Malville are different from those used at Phenix. The mechanical strength and adequate resistance of these high powered machines has yet to be confirmed regarding :

- The heat exchange capacity.
- The capacity of detecting a loss of leaktightness of S.G tubes.
- This programme will also involve a surveillance of the ageing of the structure material, the alloy 800 which constitutes one of the novel features on these steam generators.

4.2.1.4. - Handling

The handling of the sub-assemblies plays a dominant role in the avaibility factor of a power plant while ensuring flexibility in carrying out modification in the core. The follow-up of its state and of its performance involves in-situ measurements and periodic inspections. It groups the monitoring of the performance of each elementary system : the rotating transfer lock, the control station, the conditionning pit of new sub-assemblies, the irradiated fuel transfer vessel, the cooling station and the washing pit for irradiated sub-assemblies.

4.2.1.5. - Monitoring during operation

4.2.1.5.1. - Controls

Monitoring during plant operation is the object of developing specific means of testing and checking.

Accomplishing this can be of an interest which is twofold : first of all, results are obtained regarding the condition of the inspected structures and secondly, experience is acquired on the actual functionning of these specific testing and checking means. Examples can be cited : that of the MIR (Fast Reactor Inspection Machine) and the problems linked to sodium inspection.

4.2.1.5.2. - Operation monitoring

The analysis of all the observations made on the reactor whether under normal operating conditions or in the case of a malfunction or anomaly, is used to reach a number of findings and to draw conclusions from this. The thermal-hydraulic behaviour of the sodium in the large sodium plenum of the primary circuit can be cited.

Most often it involves providing feedback through the normal monitoring system of the reactor. Improvement of this monitoring will be studied either through more efficient processing of the available data or through measurement means which up until the present time have note yet been used :

- Analysis of the sodium temperature noise at the outlet of the core sub-assemblies

- The Integrated System of Delayed Neutron Detection.

4.2.2. - Research on plutonium consumption

In the field of research regarding the use of plutonium, the expected results coming from the operation of the Super-Phenix reactor are of two different sorts. They concern :

- First, a gradual conversion of the core from a fast breeder operating mode to a plutonium burning operating mode demonstrating the flexibility of the concept while providing an initial experience about the operation of a fast reactof as a plutonium consumer on an industrial scale ;

- Second, the qualification of technical solutions concerning entire fuel sub-assemblies. Such solutions have been developed within the framework of the CAPRA project in particular in order to provide decisive elements for reactor of this type.

4.2.2.1. - Evolution towards plutonium burning mode

The present core of Super-Phenix in that of a fast breeder reactor, essentially owing to the presence of fertile uranium oxide blankets which are arranged radially with 220 subassemblies and axially with uranium blanckets. The programme suggested for converting the reactor to a plutonium burner involves three phases :

- Suppression of the first row of radial blankets of the present core. Beginning with a core that produces 36 kg/TWh in total plutonium, this stage reduces production to 27 kg/TWh. On the planned cycle up to the complete burning of the present core, in other words about on year of operation, total plutonium production will be reduced by 50 kg. Evolution of the plutonium composition will be measured after unloading.

- The complete suppression of the radial blankets with the loading of core 2. Planned for the end of 1998, the loading of this core will bring about a plutonium production totalling approximately 15 kg/TWh instead of 36 kg in presence of the radial blanket. This blanket will be replaced by steel subassemblies. The neutronic characterization of this core without a radial blanket will be carried out in order to acquire unique experimental data for the qualification of computer codes on core physics.

- Specific measurements of radial distribution of the neutron flux particularly at the edge of the core, will be carried out at the loading of this core much in the same way that such measurements were carried out in 1986 at the time of the first core power increase.

- The perfecting of a core 3 globally adapted to a total plutonium consumption of about 15 to 20 kg/TWh. Such a performance will be accomplished by completely suppressing all blankets of the core, radial or axial, and by an increase of the initial plutonium enrichement.

The experimental programme associated with the planned core modifications mentionned above also involves the following measurements and examinations :

- Reactivity measurements of certain subassemblies and of the core.
- Precise determination of the power delivered by each subassembly.
- Determination of the overall core reactivity coefficients.
- Measurement of the evolution of the decay heat in the core versus time.
- Measurement of the anti-reactivity of the absorbant subassemblies.
- Special examinations after irradiation of certain core elements.

4.2.2.2. - Tests conducted on sub-assemblies

Studies carried out within the framework of the CAPRA project led to the definition of new fuel elements. They differ from those used in the breeder version in their geometry, their composition and their plutonium content is much higher.

These new fuels must be able to be used with different isotopic qualities of plutonium and possible additions of americium and neptunium.

The demonstration programme in Superphenix is aiming at an industrial pre-qualification of solutions which might be selected in CAPRA type reactors. This programme comprizes :

- At the end of 1996, the loading of two special subassemblies into the reactor characteristic of the mixed oxide solution with a high content of plutonium and corresponding dilution of the fuel. Each of these sub-assemblies has been manufactured with a plutonium having a special isotopic composition. The first sub-assembly, CAPRA 1A, contains plutonium coming from a standard PWR fuel with uranium ; the second subassembly, CAPRA 1B, is manufactured with plutonium coming from the re-processing of MOX fuel. These two sub-assemblies can be characterized basically as :

- a 31 % enrichment in plutonium
- a dilution of the fuel obtained by reducing the size of the pins, their number increasing from 271 to 397 and also by introducing among these pins a certain percentage, about one third, of pins made of steel.

- The suppression of the uranium axial blankets.

The programme might also involve :

- At the time of the core 2 loading, the introduction of special new sub-assemblies destined for a long irradiation period and having characteristics similar to those of the CAPRA 1 subassemblies :

namely

- an enrichement in plutonium of about 35 %

- a pin geometry closer to the solutions envisioned for CAPRA type reactors, taking into account present day manufacturing limitations

- one of these subassemblies will also contain an addition of neptunium.

- At the time of the core 3 loading, a zone test of several sub-assemblies representing of a core with a very high plutonium consumption.

4.2.3. - Research on the destruction of long-lived waste

- Research, especially within the framework of the SPIN programme, has revealed the particular interest that fast reactors present in destroying minor actinides, americium and neptunium, through the fission process which leads to their transformation for the most part into short-lived waste.

The long-awaited demonstrations in Superphenix have a three-fold goal : to confirm the capacity of a fast reactor to satisfy the need, to specify the performance of this solution and to provide elements indispensible for the compatibility of the two processes, plutonium consumption and actinide destruction, carried out simultaneoustly in the same reactor.

4.2.3.1. - Incineration of neptunium

The best suited solution for the incineration of neptunium, without creating great penalties, appears to be homogeneous reprocessing in fuel. The calendar of demontrations must be adapted to the possibility of having corresponding quantities of this element and also to the possibility of manufacturing the experimental elements. The proposed programme successively involves the following :

- The NACRE 1 sub-assembly in the present core standard, contains two kilograms of neptunium in a homogeneous dilution, that is to say, uniformly distributed in the fuel of the subassembly. This sub-assembly, which is to be manufactured at the COGEMA in Cadarache, will be loaded into the core in the second half of 1996 at the same time as the CAPRA 1 subassemblies. It will be unloaded with the present core, in other words, after about one year of irradiation. The expected rate of eliminated neptunium in this subassembly is about 20 % out of this irradiation time.

- Several NACRE 2 sub-assemblies. These sub-assemblies will be identical to those of NACRE 1 which means representative of the recycling in homogeneous mode. They are to be loaded in 1998 for an irradiation period lasting three years if the corresponding quantities of neptunium can be suplied within the required time.

- Possibility greater quantities in the loading of core 3, plutonium burner, under conditions defined according to results obtained from experiments carried out in the Phenix reactor and in the two preceding stages.

4.2.3.2. - Incineration of americium

In the case of americium, the situation is more complex due to the different types of formation in its major isotopes, due to difficulties in extracting it for reprocessing and also due to its gamma activity which complicates manufacturing operations. For its destruction, the choice of a heterogeneous recycling solution has been selected for the initial stage. The Super-Phenix programme has a twofold mission :

- to take benefit from the ageing of the fuel elements and thus the formation of americium 241 through the radioactive decay of plutonium in order to acquire data on the destruction of this isotope in conditions representative of the homogeneous recycling. Superphenix's present core contains an average of 0.5 % americium (mass content brought by heavy nuclei) with a maximum content slightly higher than 1 % in 5 of its subassemblies. The restart of the core will bringing precious information thanks to a certain number of examinations of these subassemblies after unloading of the core in 1998. A second stage might be decided upon preserving a few elements of the core 2 only loading them at the startup of core 3 now on the horizon for the year 2002. In this case, initial americium content of 1.3 % will be available in three assemblies.

- The realization of a few pins specially enriched in americium which may be introduced into a demonstration subassembly loaded in core 2 between 1999 and 2000.

5 - CAPRA PRELIMINARY FEASABILITIES STUDIES

5. INTRODUCTION

The aim of the first two-year phase of the CAPRA project studies (1993-1994) was to demonstrate the feasibility of a fast reactor whose net burning of Pu would be as high as possible and which could, moreover, contribute to the destruction of minor actinides. The greater part of the effort focused on a thorough study of the reference option constituted by cores using a high plutonium content oxide fuel. The potential of an alternative option involving a nitride fuel was also assessed. An exploratory investigation was made on a study of uranium-free cores allowing the highest plutonium burning rates, i.e. about 110kg/TWhe.

It is essential to stress that the work of the project was concentrated on maximum admissible Pu content for the oxide fuel, degraded isotopic quality of the reference plutonium (recycled twice in a MOX/PWR, Table 0), high power core (1500 MWe), while seeking maximal compatibility with the main options of the EFR nuclear steam supply system and with technological experience regarding fuel.

This work has been performed in the framework of the European R&D collaboration on EFR (CAPRA Project) and in close cooperation with the EFR Associates. The R&D programme in support is complemented by various international collaborations with Japan, Russia, Switzerland, Italy...

5.1. BURNER PHYSICS, GENERAL TRENDS

A parametric study carried out in order to establish the design orientations of burner cores taught us first that a considerable reduction of the fuel inventory (or 'dilution') is always necessary to be able to operate a large core with a high plutonium content and therefore with an attractive plutonium burning performance. This dilution results in a decrease in in-pile fuel residence time as well as in a reduction (favourable) of the sodium void reactivity, whereas a decrease of the uranium content of the fuel brings about a reduction of the Doppler effect, a decrease of the conversion ratio which causes a daily reactivity loss that makes it difficult to achieve long irradiation cycles, as well as a reduction of the delayed neutron fraction.

It was demonstrated that cautiously resorting to poisoning, in which the excess reactivity linked to the increase of the fuel plutonium content is compensated by the introduction of a neutron absorber, was a possibility and allowed certain of the drawbacks involved in the dilution process (short in-pile fuel residence time, for example) to be slightly reduced at the expense of a deterioration of the sodium void and Doppler effects.

The management of the marked reactivity loss and the optimization of the relative values of the Doppler and sodium void effects appear to be the two major issues to be dealt with so as to ensure the feasibility of Pu burner cores. In this respect, heterogeneous core designs, in which a part of the inert material replacing the fuel in the dilution process is gathered in specific subassemblies, are of particular interest.

5.2. OXIDE REFERENCE OPTION

5.2.1. Justification of the Choices

The idea was, of course, to take the use of oxide fuel to its limits whilst remaining sufficiently 'conventional' to benefit from the considerable experience that is available concerning this fuel (the choice of fuel pellet technology for example).

The search for compatibility with present-day fuel-cycle technology (dry route fabrication, PUREX process reprocessing) led to the choice of a maximal plutonium content of 45%, corresponding to a Pu burning rate of the order of 70kg/TWhe. The thermodynamic solubility of a perfectly homogeneous 45% Pu content oxide was demonstrated by means of new experimental results although the kinetics of dissolution of those fresh pellets was slow. A first stage in the qualification of the 45% Pu content choice has therefore been carried out, knowing that the results obtained with dry route pellets of slightly lower plutonium content showed that in the event of major difficulties a return to a maximal Pu content of about 40% could possibly be envisaged later, at the expense of a loss of 5 to 10kg/TWhe on the Pu burning rate.

The implementation of 45% Pu oxide in a 1500 MWe core implies a reduction by about a factor of 2 of the fuel inventory (total heavy atoms) as compared to a fast breeder core of equivalent size and power. In the "4/94" promoted design, this is done by pins of small diameter containing oxide pellets with a large central hole, by heterogeneous subassemblies comprising a large number of pins (469) about one third of which would be empty of fuel but filled with an inert material (the best candidate appears to be spinel, $MgAl_2O_4$), and by the presence of around fifty diluent subassemblies containing no fuel. These diluent subassemblies, which constitutes an essential particularity of the CAPRA design, were accommodated in a core whose volume was that of the EFR reference core by designing a subassembly with reduced pitch (Figure 6 and Table 1).

An irradiation programme designed to validate and optimize these main fuel choices was defined and launched in 1994. The first results on irradiated 45% Pu oxide, implemented in the form of small pellets with a large central hole, are expected in 1995 (IFOP1 irradiation in SILOE,

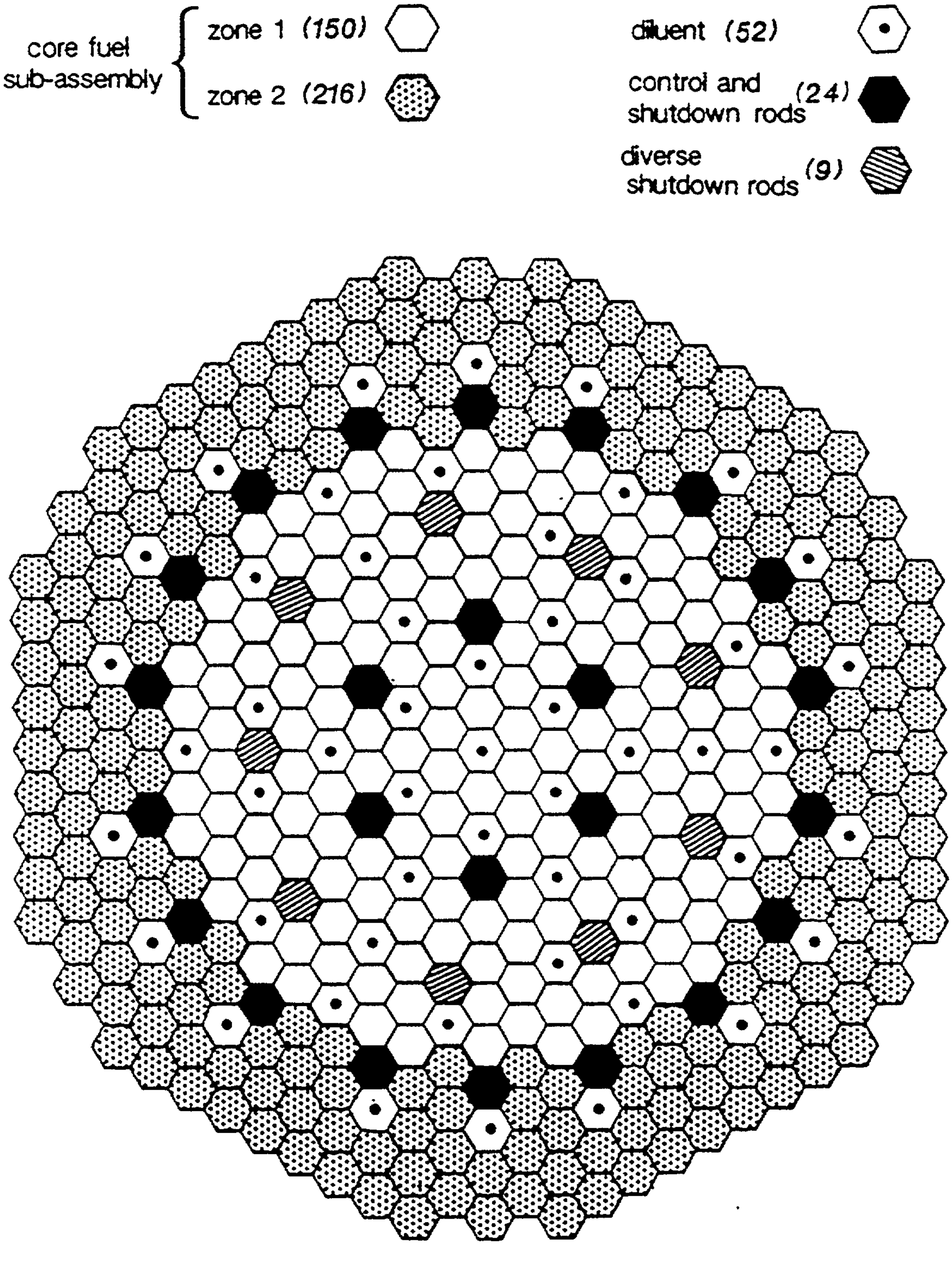


FIG. 6. Layout of the reference CAPRA oxide core (4/94).

already achieved, and TRABANT in HFR). The irradiation programme hinges on the use of the PHENIX reactor to study the behaviour of individual pins and to optimize the design of the CAPRA pins (CAPRIX 1 irradiation, then CAPRIX 1B, 2... for the fuel pins and MATINA for pins containing inert material), and on SUPERPHENIX to study the behaviour of a heterogeneous bundle (CAPRA 1 subassemblies placed under irradiation in core 1) in a procedure whose goal is the irradiation of a significant number of representative CAPRA core subassemblies in the SUPERPHENIX core 3, by the year 2000.

5.2.2. Core Evaluation

The "4/94" oxide core has been the subject of a detailed evaluation in the fields of neutronics, safety and the behaviour of the fuel. Its main results as regards performance are recalled in Table 2. The plutonium burning is in agreement with the target aimed at (70 to 80 kg/TWhe). The production of minor actinides may seem to be rather high: it is directly linked to the degraded quality of the plutonium assumed in the study and, in fact, for an equivalent Pu vector in the fuel, the EFR and the CAPRA cores shown in Table 2 have comparable performances. This core appears to be viable as a whole, at least within the limits of the characterizations carried out so far which are, however, of relatively exhaustive character. This evaluation shows:

- Strong points fundamentally linked to the choice of a heterogeneous core design: better response to whole core flow transients because of the significantly improved sodium void effect; more generally, the response of the core to design-basis accidents (pump rundown, control-rod withdrawal, LIPOSO failure, etc...) has been assessed and shown to be very similar to that of the EFR core or slightly improved; for beyond-design-basis events, a preliminary assessment of the response to core-disruptive accidents suggests that it will be slightly improved (initiation phase) over that of EFR; flexibility presented by this design, notably for the use of plutonium of variable quality and the introduction of minor actinides

- Weak points: firstly the management of the high reactivity of the core that requires temporary poisoning of about half of the diluent subassemblies which are replaced at mid-cycle, and secondly the reduction in the in-pile residence time of the fuel (mainly imposed by the target burn-up which, in the framework of the study has remained that of the EFR project, namely 20 at%); the question of recriticality after CDA and local melting events remains to be addressed in depth; while there is an expected tendency for larger reactivity additions than for EFR, the flexibility of the heterogeneous layout allows the possibility of design features to ameliorate the response; further studies of this question are necessary;

- A few uncertainties mainly related to the choice of fuel with a large central hole and to the heterogeneity in the bundle and in the core.

To widen the options, some prospects of progress and/or alternatives have been identified and some of these have already been evaluated.

The neutronic validation of the CAPRA oxide core options and, in particular, the confirmation of the evaluation of their reactivity effects (sodium void, heterogeneity inside the S/A, critical balance, ...) and delayed neutron fraction is the object of the CIRANO programme, lasting two to three more years, in the experimental reactor MASURCA at Cadarache. After its first feedback expected by the end of 1995 when a restricted oxide core configuration will be achieved with a high Pu content (55%) and degraded Pu, a second phase of the programme will focus on the study of the heterogeneity effects (diluent subassemblies and "empty" pins in the fuel bundle) and the impact of the introduction of new materials.

5.2.3. Capacity for Minor Actinide Destruction

The procedure consisted in applying the main recommendations to date extracted from the SPIN programme to the reference CAPRA core: homogeneous recycling of Np and heterogeneous recycling of Am.

Studies concerning the homogeneous recycling of Np showed that up to 3 to 5% Np can be introduced into the oxide, while maintaining a (Pu+Np)/(U+Pu+Np) ratio of 45% since vis-à-vis reprocessing, Np behaviour would appear to be similar to that of Pu. The core thus contains about one ton of Np whose transmutation rate is about 50% for the approximately 900-efpd total fuel irradiation. In these conditions, plutonium burning of course decreases whereas the overall burning of actinides (Pu+Np) remains essentially the same. The known consequences of the introduction of Np on the Doppler and sodium void effects (even if limited: ~10% increase in sodium void reactivity and ~10% decrease in Doppler constant for 1% of Np in heavy metals) have to be remedied by the insertion of a moderator. The heterogeneous design is turned to good account by inserting $^{11}B_4C$ into the empty pins as a replacement for the inert material, thus allowing satisfactory reactivity effects to be regained.

The heterogeneous recycling of Am in the form of target subassemblies arranged in a first peripheral ring around the core (about 1.5t of Am) only slightly alters the core performance. The irradiation of the targets is limited to 9 cycles in order to not exceed either a dose of 200DpaNRT on the cladding or the power that would be released by blanket subassemblies in a conventional breeder core. In these conditions, about 75% of the initial Am is transmuted, 50% of which is changed into Pu and short-lived Cm which will decay to Pu.

The qualification of the homogeneous Np recycling mode is in particular the object of several experimental irradiations: the TRABANT experiment in HFR (high Pu and Np contents), SUPERFACT2 in PHENIX (high burn-ups on Np and Am doped oxide fuels) and the whole subassembly NACRE1 irradiation in SUPERPHENIX. Am target irradiation is foreseen both in SUPERPHENIX (core 2 or 3) and in the frame of the European EFFTRA collaboration programme.

5.2.4. Flexibility to Plutonium Quality

The chosen core design is able to deal with a large range of civil Pu qualities. Without having to unduly alter the geometric characteristics of the subassembly and the fuel pin, it has been demonstrated by current studies that the CAPRA reference core can accommodate:

- More degraded plutonium (for example corresponding to multi-recycling of the reference plutonium in the CAPRA reactor, Table 3) by increasing the fuel inventory replacing the empty pins with fissile pins; moreover, the negative impact of degraded Pu on the sodium void effect can be remedied by the introduction of a moderator such as $^{11}B_4C$ into the remaining empty pins; a particularly attractive feature is the limited increase of minor actinide production in multi-recycling: it only rises to about 12kg/TWhe (reactor balance) for the equilibrium Pu given in Table 3;

- More reactive plutonium (for example standard PWR plutonium) that can be handled either only slightly decreasing the fuel enrichment, thus accepting a small decrease of the Pu consumption level, or, while keeping constant the fuel Pu content and therefore the burning capability (about 70 kg/TWhe), by making a further step in the dilution process by adding some Ceria (CeO_2) in the oxide solid solution.

Using weapons-grade plutonium can also be envisaged. Its mixture with a degraded plutonium poses no problem of accommodation and takes us back to our previous remarks; its non-'diluted' use is possible either accepting a decrease of the fuel enrichment and therefore also of the Pu consumption level to about 50 kg/TWhe, or, if one was to keep the objective of about 70 kg/TWhe Pu burning, changing to a ternary fuel of $(U,Pu,Ce)O_2$ type that would allow a further important step in the dilution while keeping the conventional pelletized fuel.

5.2.5. Integration of the Core into the Nuclear Steam Supply System

No major difficulty is to be anticipated in the integration of the reference oxide core into a nuclear steam supply system of EFR type. Certain minor geometrical adaptations are necessary because of the slight reduction in the radial dimensions of the subassemblies (core cover plug, diagrid). However, the larger number of subassemblies in the CAPRA core (418, including the diluents, compared to 388 for the EFR Consistent Design) can be accommodated on the diagrid without the need for changing its diameter, and, therefore, without any consequences on the reactor vessel dimensions.

However, the shortened cycle time and the need to handle diluent subassemblies at mid-cycle require improvements of the fuel handling system performance in order to keep the load factor to an acceptable level: in-gas handling route, doubling of the washing pits, reduction of the handling-to-'on-power' and 'on-power'-to-handling switching times. With these improvements, it is possible to achieve the objective of 33 days for the mean annual duration of the scheduled outages, compliant with an 80% load factor requirement in the EFR project specifications.

5.2.6. Reversibility

Reversibility is understood here as the capacity of a reactor design to be able to operate in a wide range of in-situ fuel conversion ratios, from distinctly burner situations right up to breeding. Total reversibility with the reference features of the EFR Consistent Design is possible if the objectives in terms of Pu burning remain limited to 30kg/TWhe. It can be increased, as shown in the upper part of Table 4, up to about 50kg/TWhe if one considers the first step towards CAPRA type designs: the introduction of fuel-free pins and a change to a greater number of pins in the fuel subassembly (469).

In the case of a CAPRA design characterized by a reduction of the subassembly pitch (more ambitious burning rate and optimized core parameters for this burning rate), the return to breeding is entirely possible, either by reverting to the EFR reference system if the modification of the subassembly pitch with its implications on the reactor structure (diagrid, core cover plug) is acceptable, or by keeping the CAPRA subassembly design (a compact version of the EFR called CAPRA Breeder, has therefore been characterized; see lower part of Table 4), at the expense of drawbacks which are very slight as regards the breeding rate and the in-pile fuel residence time.

5.2.7. Costs

In view of the small impact of the integration of the CAPRA core into the EFR nuclear power supply system and of the measures taken to conserve a satisfactory load factor, it emerges that the investment and operational costs remain, at a first estimate, comparable to those of the EFR reference power plant. However, there is a significant impact of both the short in-pile fuel residence time (doubling of the S/A throughputs in the cycle) and of the more sophisticated S/A design on the cost of the fuel cycle. As the cycle cost contributes about 10% of the electricity generating cost, the cost of the kWh produced by a CAPRA-type Pu burner will exceed that of an EFR by a few percent. The high burn-up core version of the "4/94" design, or the poisoned type cores considered in the early stage of our studies, although they are associated with an important reduction of flexibility, show a much more favourable figure. There remains to be assessed, of course, the cost of managing the additional structural waste due to the special design of the CAPRA subassemblies, to be associated with the increase of subassembly throughput, which should be weighed up against the benefits of the reactor in terms of plutonium stockpile limitation.

5.2.8. Conclusion

At this stage, one can conclude that the proposed burner-core design seems to be viable as a whole. Its strength lies in its very great flexibility, resulting from the heterogeneous character of the bundle and the core which allows minor actinides to be easily taken into account and makes multiple plutonium recycling quite conceivable. The technology implemented is rather conventional and makes it possible to go back to lower Pu burning situations or even to breeder situations at the expense of drawbacks that remain very limited. Nevertheless, when going to burning, an increase of the fuel cycle cost looks unavoidable.

5.3. ADVANCED OPTIONS

5.3.1. Nitride

It is undesirable to put forward a new fuel resulting in a new high-level activity waste, in this case C14 produced by the neutronic reactions on the nitrogen constituent of the nitride. The solution lies in the enrichment of natural nitrogen in N15. An evaluation of the technico-economical feasibility of enrichment of nitrogen has therefore been carried out. It showed that the process is technically conceivable for the additional fabrication cost of a nitride subassembly of the order of 20%. This additional cost should, of course, be weighed up against the benefits to be otherwise expected from the use of nitride.

These benefits, particularly in a CAPRA application, have not yet been definitively defined: in addition to the traditional interest of nitride (compatibility with sodium, good thermal conductivity and higher fuel density), this fuel would have the advantage of being in principle soluble in nitric acid, and therefore compatible with the PUREX reprocessing, in the entire Pu content range. This point has not yet been definitely demonstrated and studies on the solubility of nitrides with a high Pu content, initiated in 1994, are to continue in 1995.

The interest of nitride in the frame of CAPRA would be threefold: as an alternative to oxide; as a solution allowing access to an intermediate Pu burning range between the reference oxide and the cores based on an uranium-free fuel; and, finally as a Pu compound for uranium-free fuels. At present, the last of these seems most promising.

5.3.2. Pu without U

The studies carried out in this field have been exploratory in character insofar as questions of principle had to be answered concerning the safety of these cores, a priori characterized by the absence of Doppler effect, before starting more detailed evaluations. The most notable result is that it is quite clear that there is no impossibility as regards safety (stability, flow transients, reactivity transients, core accidents) provided that a fuel design, meeting a certain number of constraints, can actually be put forward. At the present stage of the studies, some promising generic fuel families have been proposed but no clear and definitive choice has emerged, since a specific experimental validation programme is needed.

One important lesson drawn from the studies was that the low Doppler effect does not pose any problem for flow transients insofar as it is compensated by the low sodium void effect resulting from the very high dilution of the uranium-free fuel in the core. However, to achieve acceptable behaviour in the reactivity transients, besides requiring fuels with considerable margins as regards melting, it seems indispensable to find another prompt negative reactivity feedback. The studies showed that, to a certain degree, axial thermal expansion of the fuel could ensure this function, or that certain neutron absorber materials, such as W for example, could restore a Doppler effect but important uncertainties remain in the estimation of its absolute value. The studies revealed that these absorber materials have, as expected, a very negative impact on the sodium void reactivity and that only materials with a moderate neutron absorption cross section could be tolerated. For these two situations, the requirement was to find a fuel solution in which any power rise would instantaneously put either the expansion or the Doppler reactivity effect into action. This supposes excellent thermal coupling between the Pu and the other constituent elements of the fuel. For the expansion effect, moreover, the design of the fuel must allow the latter to expand freely within its cladding.

In a process for which the implementation of a structural Doppler effect is desired, massive introduction of an absorber structural element would result in considerable increase of the positive core sodium void reactivity: to restore margins as regards this value, it seems advisable to consider smaller sized cores. As a consequence, two possible routes for further investigation (they can be mixed) seem to be either a large core in which an efficient axial expansion of the fuel is required or a smaller core in which a structural Doppler effect is sough.

5.4. GENERAL CONCLUSION

5.4.1. Feasibility

The feasibility of a Pu burner reactor based on the use of oxide fuel has been demonstrated. Its limits in terms of Pu burning are 70 to 80kg/TWhe corresponding to the choice of a maximal fuel Pu content of 45%. This feasibility was proved by considering the limit case of a high power core (1500MWe) for which there emerged acceptable overall behaviour associated with very great flexibility, either vis-à-vis the nature and quality of the actinides to be processed or the possibility of regulating the Pu conversion or breeding ratios. The impact on the cost of the kWhe should be limited, although an increase of the cycle cost is unavoidable.

The feasibility of reactors based on the use of uranium-free fuel has not yet been fully established. The exploratory studies that have been performed have shown that there is, a priori, no obvious killing argument, and that this path should be investigated further. The challenge is still, more than ever, to find a fuel that can meet the required design constraints. Unlike the case of the cores implementing oxide, recourse to cores of reduced power cannot be entirely ruled out (improved reactivity coefficients, safety behaviour,..).

5.4.2. The CAPRA Contribution vis à vis the Cycle

The relevance of a CAPRA-type reactor is that it can be integrated into the existing nuclear park to regulate (stabilize) Pu stocks whilst exploiting this element energy worth, and to reduce the long-term potential radiotoxicity source term of final wastes. As regards these two objectives, and as part of a procedure which enables the use of existing PWR power plants to implement the first steps of plutonium recycling (one or two recycles) prior to considering Pu or Pu+Np multi-recycling in the burner (an operation whose feasibility has been demonstrated), one oxide CAPRA reactor for 4 PWR reactors would allow the stocks of Pu and of Pu+Np to be stabilized. This operation would be accompanied by a quite considerable reduction of the actinide production (Pu+Np+Am+Cm) as compared to the open cycle and with respect to concievable multirecycling of MOX in PWRs. The first objective is thus amply met. As regards the reduction in long-term radiotoxicity, the multi-recycling of Pu+Np in CAPRA reactors, along with their load of target Am, would be expressed by an already highly significant gain of about a factor of 10. This factor can be increased up to 30 to 40 if the Am production is stabilized : this target seems to be achievable in the CAPRA reactor, by specific and optimized design and management of the Am targets that need to be studied in the next phase of work.

Table 0: reference plutonium quality used in the CAPRA studies

Pu238	Pu239	Pu240	Pu241	Pu242	Am241
5.6 %	39.1 %	26.7 %	13.0 %	14.3 %	1.3 %

Table 1 : main characteristics of the reference CAPRA oxide core

	CAPRA core	EFR CD/91
Number of fissile S/As	366	387
Number of diluent S/As	52	1 (central position)
Number of control S/As	24	24
Number of complementary shutdown S/As	9	9
S/A pitch (mm)	181.4	188
External across flats distance (mm)	176.4	183
Internal across flats distance (mm)	167.6	174.2
Number of pins per S/A	469	331
Number of fissile pins per S/As	336	331
Pin length (mm)	2305	2645
Fissile column length (mm)	1000	1000
Spacer wire helicoidal pitch (mm)	150	180
Spacer wire diameter (mm)	1.2	1.14
External clad diameter (mm)	6.35	8.2
Internal clad diameter (mm)	5.45	7.16
Fuel pellets diameter (mm)	5.27	6.94
Fuel central hole diameter (mm)	2.16	2

6 - EUROPEAN FAST REACTOR DESIGN

6.1. Introduction

The role for Fast Reactors has always been to secure the long term fuel supply for electricity generation with its ability to produce over 50 times more power from the uranium fuel than the thermal reactors. This remains the ultimate goal but in the meantime the uranium supply remains secure and economic for the thermal reactors and fast reactors are being evaluated as an economic and flexible machine to consume the plutonium and transmute minor actinides coming from the thermal reactors.

It is against this evolving background that the European Fast Reactor (EFR) project was launched in 1988. It has now reached an important step with the completion of the Concept Validation Phase. The twin goals that the customer utilities set at the outset for the design have been demonstrated to be achievable:

- economic performance with the generating cost of a commercial series of EFR competitive with its contemporary PWRs,and
- licensable in all participating countries with a requirement for a safety level which meets the ambitious targets of future nuclear plants.

The R&D support has been extensive and has provided comprehensive validation of the design features necessary to meet these goals.

Table 2 : main operational features of the reference CAPRA oxide core

	CAPRA core	EFR type CD/91*
Frequency	3	5****
Cycle length (day)	285 but with some diluent S/As handling at mid-cycle	340****
Fuel Pu content (% in mass of HM)	44	21****
Pu burning (kg/TWhe**)	74	-8 (+16 core; -24 blankets)
Minor actinides production Np/Am/Cm (kg/TWhe**)	9.7	2.6*****
Peak burn-up (% h.a)	20	20****
Peak damage (Dpa NRT Fe)	124	190****
Reactivity loss over the cycle (pcm)	9000	2900****
Doppler constant at End of Cycle (pcm)***	-455 (core)	-650****(core) -140****(blankets)
Sodium void worth at End of Cycle (pcm)***	1560	2100****
Peak linear heat rating (W/cm)	500	480 500 (w/o fertiles)
Nominal clad temperature (°C)	630	630

* : with bankets and using a standard PWR plutonium (PWR33)

** : thermodynamic yield of 0.405, reactor balances

*** : including for the CAPRA result an heterogeneity correction

**** : taken from EFR-core data file

***** : standard PWR plutonium

6.2. The EFR Programme

By March 1988, in response to the European Fast Reactor Utilities Group (EFRUG) request, the design companies were collaborating as EFR Associates and had agreed a work programme with EFRUG. The allocation of responsibilities was established and a design team totalling some 250 engineers commenced work on the European Fast Reactor.

In June 1990, the organisation was centralised with the establishment of the Project Management Team (PMT) in Lyon to coordinate the design work being carried out at the three centres of Bergisch-Gladbach (SIEMENS), Lyon (FRAMATOME) and Risley (NNC). The PMT comprises engineers delegated from the three companies participating in EFR Associates. In addition, the progressive re-alignment of the R&D programmes at the European research centres of CEA, SIEMENS (formerly INTERATOM), KfK and AEA to meet the needs of EFR, was completed.

Table 3 : equilibrium plutonium composition in the multi-recycling process in a CAPRA reactor

Pu238	Pu239	Pu240	Pu241	Pu242
3.4%	32.9%	40.1%	8.1%	15.5%

Table 4 : illustration of Fast Reactor reversibility

EFR - CD Reversibility S/A Pitch : 188 mm				
	EFR - CD	→	EFR - CD Burners*	
Fuel pellet geometry (mm)	6.94x2.0	→	6.28x2.11	5.72x2.35
Number of pins per S/A (Fissile/Total)	331/331	→	331/397	331/469
Fuel residence time (efpd)	5x340	→	5x270	5x210
Mean fuel Pu content (%)	21	→	28.5	35
Breeding Gain range	-0.2 +0.15	→	-0.36	-0.49
Pu burning (kg/TWhe)	-	→	40	54
CAPRA "4/94" Reversibility S/A Pitch : 181.4 mm				
	CAPRA Breeder*	←	CAPRA "4/94"	
Fuel pellet geometry (mm)	6.52x2	←	5.27x2.16	
Number of pins per S/A (Fissile/Total)	331/331	←	336/469	
Fuel residence time (efpd)	5x320	←	3x285	
Mean fuel Pu content (%)	22.2	←	43	
Breeding Gain range	-0.23 +0.12	←	-0.65	
Pu burning (kg/TWhe)	-	←	74	

* : simplified and non-optimized calculations

The goal is to have a proven fast reactor design available for commercial deployment in the first decades of the next century, when the need for replacement of ageing nuclear power plants arises in Europe. A necessary prerequisite is to have operational experience from an EFR demonstration plant in advance of commercial deployment. To achieve this goal, the EFR programme shown on Figure 7 has been structured into the following main phases.

Conceptual Design Phase March 1988 - March 1990

The first two year programme of work was directed at assessing and gradually eliminating various design options until the Conceptual Design had been identified. An initial statement

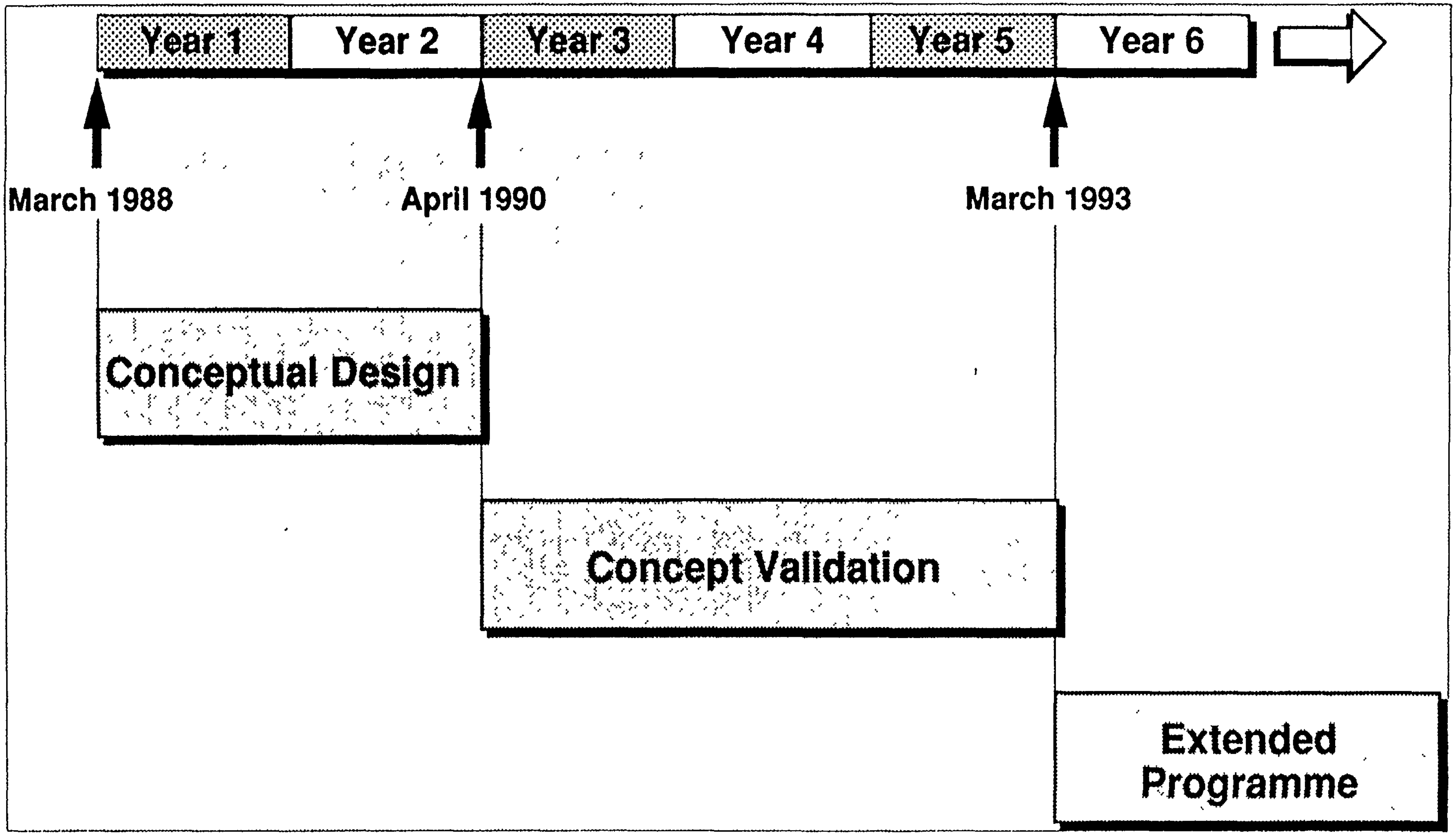


FIG. 7. EFR Programme.

outlining the essential elements of the safety case was produced and the R&D needs were identified. These relate to the validation of the compact reactor design of EFR, to the development of advanced components offering cost saving potential and to the substantiation of an up-to-date safety approach.

The aim was to combine the best features of previous designs (RNR 1500 in France, SNR 2 in Germany and CDFR in the UK) with those innovative features which offered benefits in terms of safety, reliability and cost.

Concept Validation Phase April 1990 - March 1993

In this three year period following on from the conceptual design, the system engineering for EFR was completed and the R&D results were integrated into the design. Detailed studies concentrated on the analysis of crucial features of the design, on comprehensive assessments of the safety approach and the economic prospects.

In advance of any formal involvement of the national regulators, EFRUG decided to seek the advice from a representative panel of assessors composed of senior safety experts from various national organisations.

The economic assessment was a joint activity between EFRUG and EFR Associates. The assessment involved a number of qualified nuclear equipment manufacturers in Europe who provided quotations for the main components. Also, the major companies for fuel fabrication and reprocessing provided fuel service cost information enabling an assessment of the fuel cycle costs to be carried out.

The output from this phase is a technically and economically well established Nuclear Island design and a preliminary non-site specific safety analysis report, accompanied by initial probabilistic risk assessment studies.

Extended Programme since April 1993

The technical and economic achievements of the Concept Validation phase satisfy the objectives set by EFRUG. But before commencement of the preconstruction phase it has always been required to have the essential feedback of a period of successful operation of Superph_nix. Because of the possible consequences on the design it is also important to have completed the studies which establish the range of flexibility of FR to respond to the prevailing needs of fuel cycle, in particular the potential to manage the plutonium stockpile arising from the thermal reactors and transmute minor actinides in order to reduce the quantity of high level waste. Therefore EFRUG decided to delay any decision for entering a cosntruction phase. Advantage will be taken of this time to investigate further economic improvements to the inspectability and safety of the design.

6.3. Robust Design

Nuclear Island

The centre of the Nuclear Island (NI) is formed by the cylindrical Reactor Building (RB) with three adjacent Steam Generator Buildings (SGB). In addition the NI incorporates the Switchgear Buildings which house the essential and non-essential electrics and the main control room, and the Auxiliary Building housing the fuel and component handling equipment, decontamination facilities and stores for new and spent fuel (see Fig.8 and 9).

The turbine house is located opposite the auxiliary building and connected to the SGBs by feedwater and main steam lines. Tunnels are used for cables and pipes to connect the different buildings.

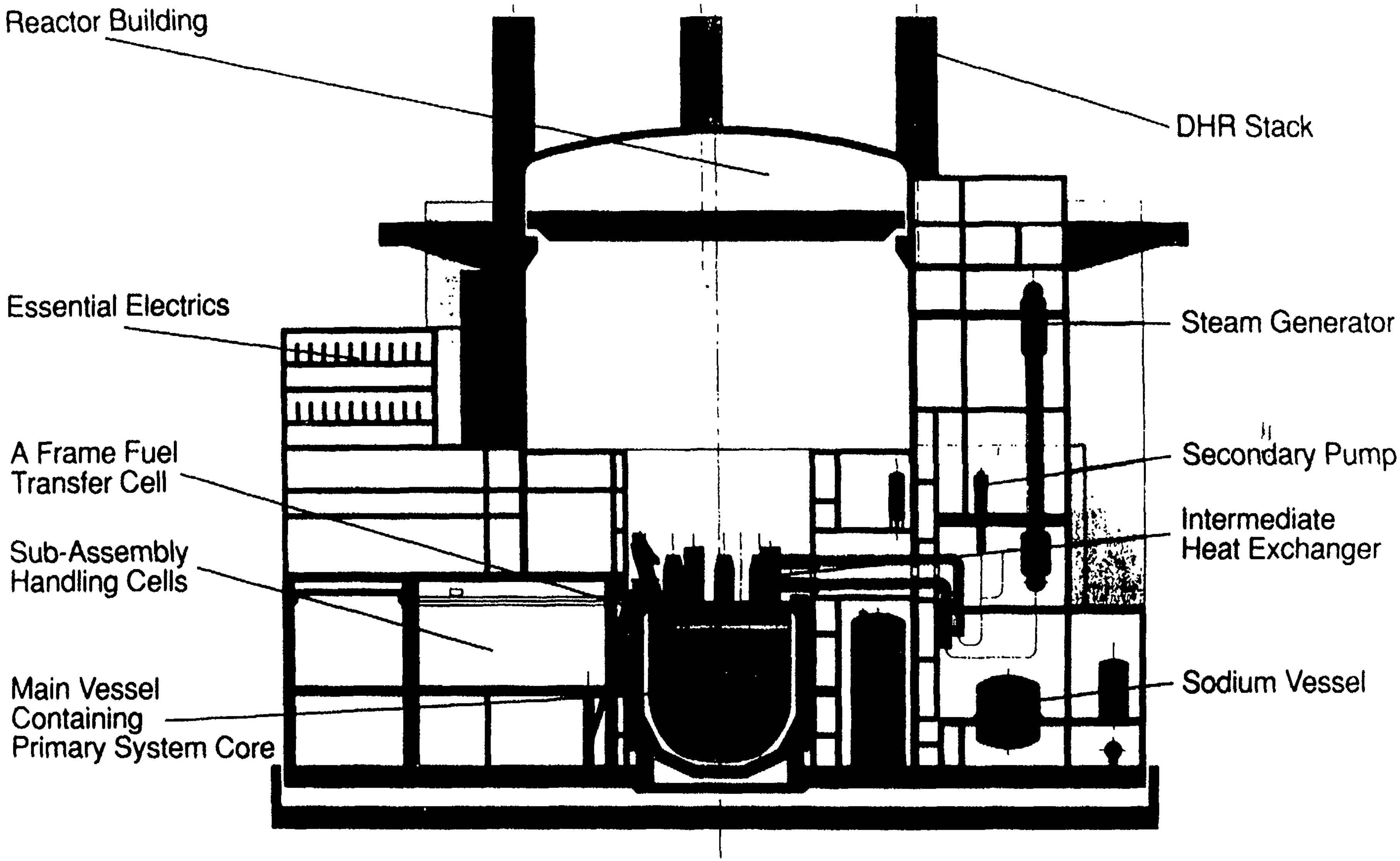


FIG.8. EFR nuclear island - elevation.

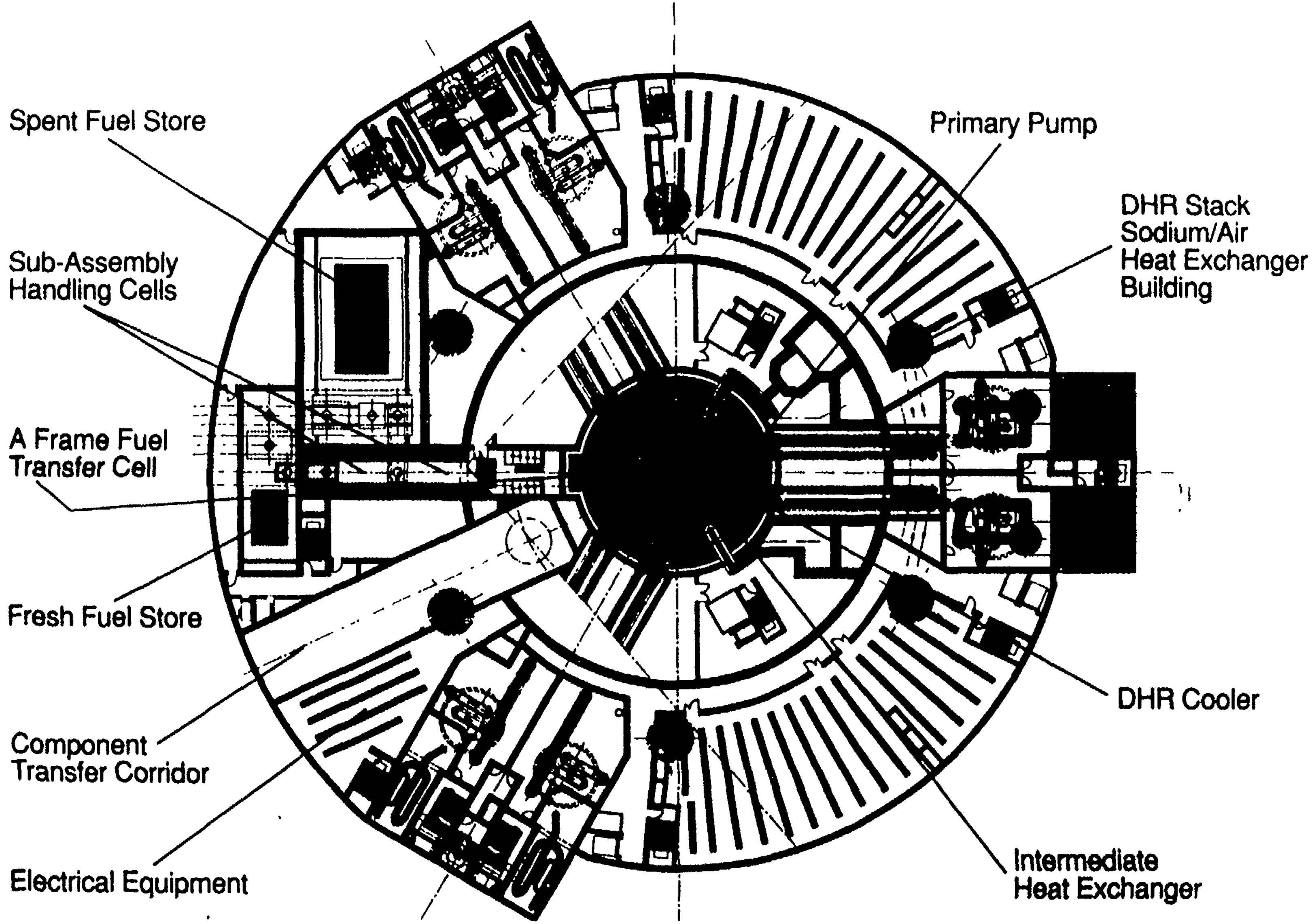


FIG. 9. EFR nuclear island - plan view.

The Reactor Building is designed to accommodate the reactor and its associated protection and cooling systems based on a six circuit sodium cooling system for heat transfer to the steam generators.The building is constructed of unlined reinforced concrete and forms the secondary containment boundary. It is designed to prevent the release of radioactivity, to withstand the pressure resulting from a sodium fire and to provide radiological shielding.

Other major features for sizing the Reactor Building are the provision for flasking of reactor components via a polar crane and a transfer corridor, fuel handling by an A-frame and via a handling cell, a three safety divisions/two conventional sections-electrical system and provision for protection against aircraft crash.

The Reactor Building together with the adjacent Steam Generator Buildings, Switchgear Buildings and Auxiliary Building are all on a common raft with bearing pads for effective isolation of horizontal earthquake-induced loads.

Secondary sodium circuit are employed to separate the steam plant from the core. It means that because of the low activity of the secondary sodium that the dose to operators is very low. The steam generators and decay heat removal circuits are arranged in pairs on a simple radial layout.

The leading parameters are given in the following table:

Leading station performance parameters.

Reactor heat output	3600 MWt
Alternator output	1550 MWe
Nett electrical output	1450 MWe
Nett efficiency	0.4
Core inlet temperature	395°C
Core outlet temperature	545°C
Feedwater temperature	240°C
Steam temperature	490°C
Steam pressure (at SGU)	185 bar
Primary pump flow rate	20172 kgs-1
Primary circuit pressure drop	600 kPa
Secondary pump flow rate	15330 kgs-1
Secondary circuit pressure drop	400 kPa
Fuel linear rating (nominal max SOL)	520 Wcm-1
Fuel linear rating (nominal max EOL)	410 Wcm-1
Fuel clad temperature (nominal max)	645°C
Maximum burn up (target value)	20 at.%

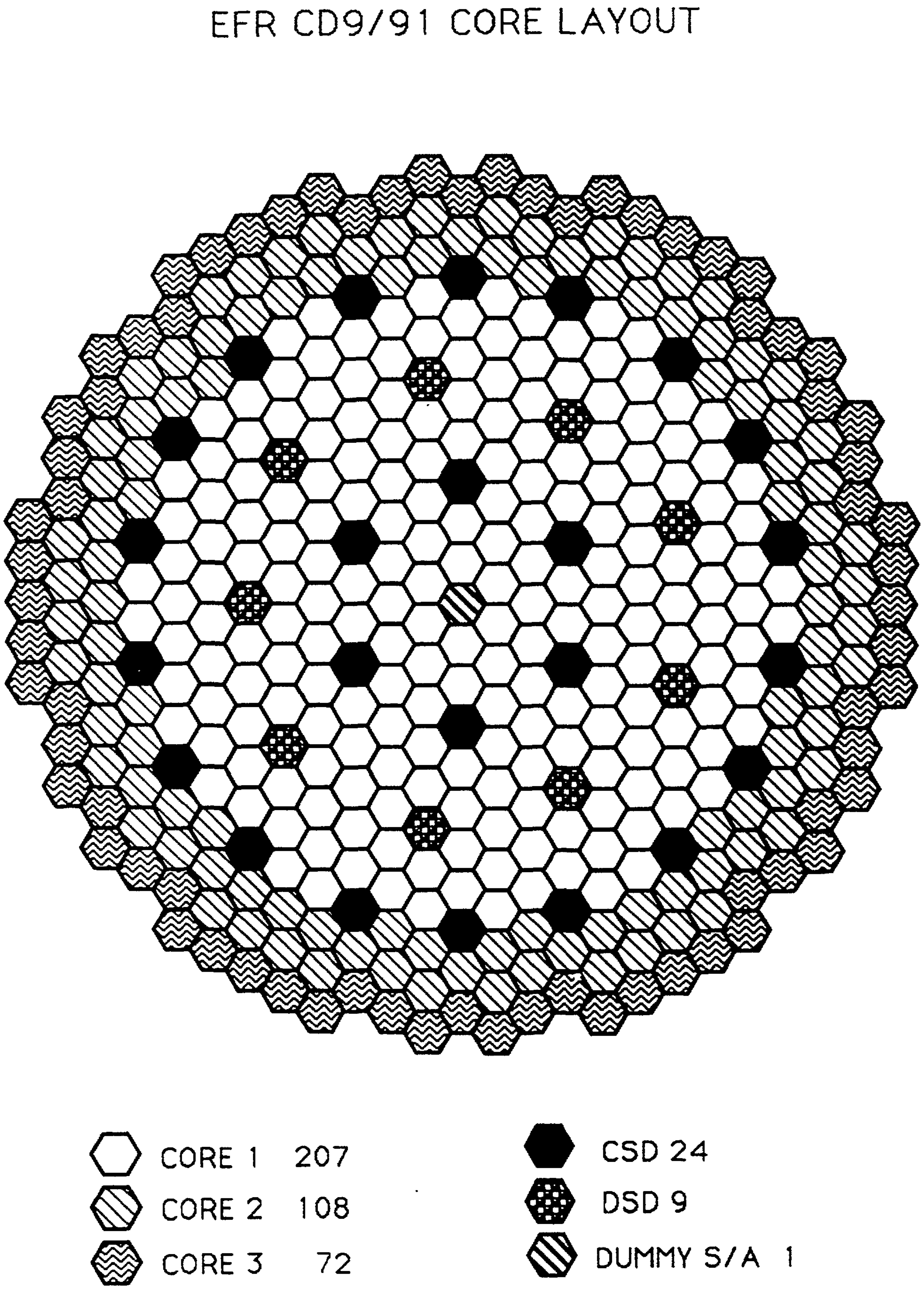


FIG.10. Core layout.

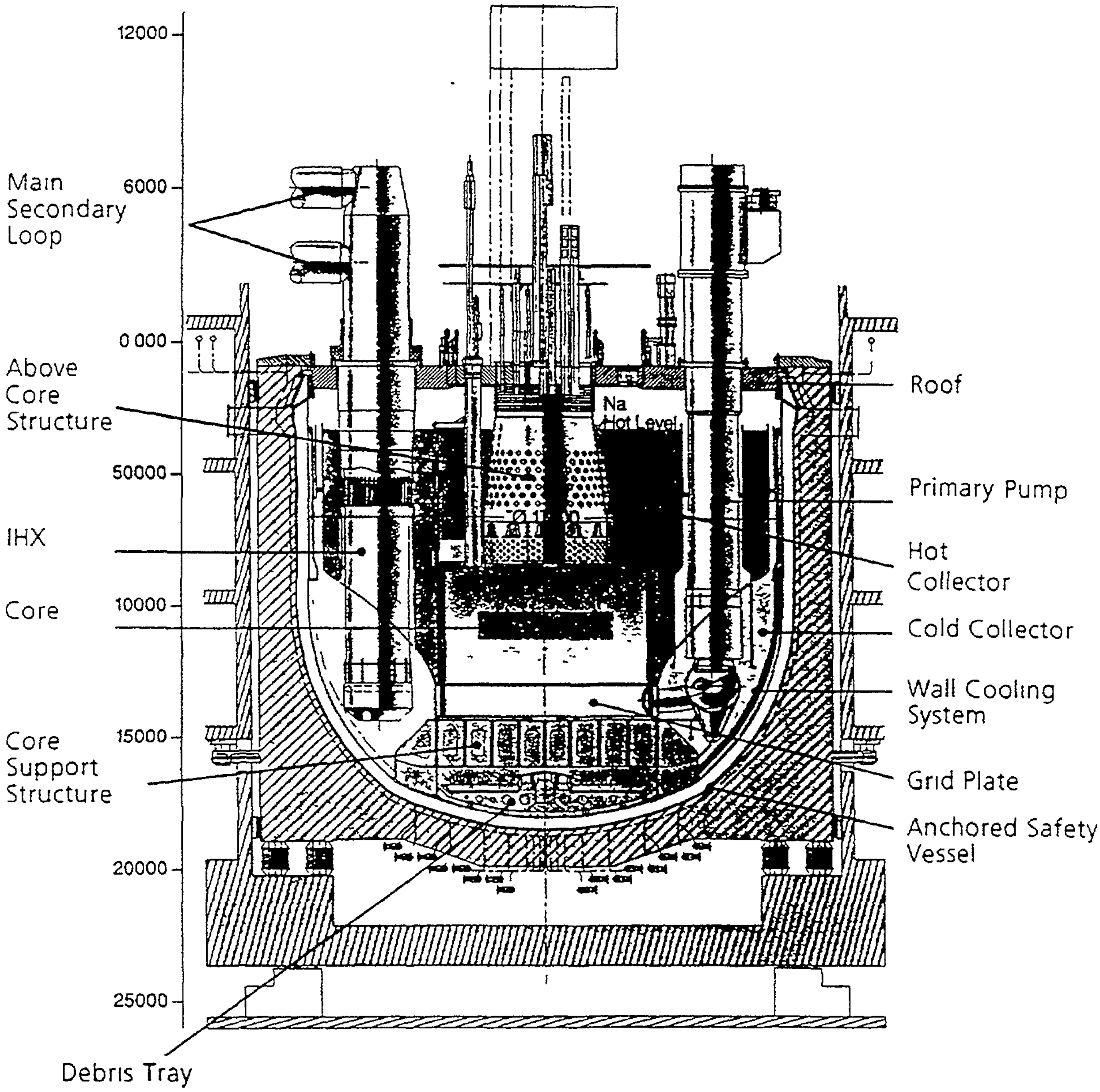


FIG.11. EFR reactor unit.

Reactor Core

The EFR core has been designed to have a high burn-up (20% heavy atoms) and long residence time in order to minimise the fuel cycle cost and reactor outage time for refuelling.

Two design options are considered for the reactor core: a conventional homogeneous core and an axial heterogeneous core with an internal fertile slice just below core midplane. Both concepts (see Fig10) have three core zones with different plutonium contents and are fully compatible with each other. The core is surrounded by one row of breeder subassemblies and an axial breeder blanket located above and below core.

A feature of the design flexibility with regard to the breeding characteristics, either the suppression of the axial and radial blanket, or the addition of axial blanket (up to a total of 0.8 m) and of one additional radial breeder row are possible giving a range of breeding gain from -0,2 to +0,15.

Reactor Unit

A compact reactor unit(see Fig.11)has been achieved with considerable simplification to the structures, components and improved surveillance potential. The large pool plant layout is an evolution from Superphénix taking full advantage of the national studies for CDFR, SNR 2 and Superphénix 2.The core, neutron shield and internal fuel store are supported by a diagrid which sits on a strongback to transfer the weight to the main vessel.

The main vessel also forms part of the primary containment along with the upper closure comprising the reactor roof, two rotating plugs and the component plugs located in the roof penetrations. The upper closure provides the thermal and biological shielding and is separated from the hot pool by an argon cover gas and insulation on the downward facing surfaces.

Sodium is circulated through the core by 3 primary pumps and the heat transferred to the secondary sodium by 6 intermediate heat exchangers (IHXs). Hot sodium leaving the core is separated from the cold sodium feeding the core by a single shell called the redan.

Decay heat can be rejected from the hot pool by the 6 coolers which form part of the direct reactor cooling (DRC) system.

A key to the compactness of the design is the reduction in the rotating plug diameter, made possible by the fuel handling arrangements which employ an intermediate 'put-down' for the fuel subassemblies in the central part of the core.

A smaller number of the large components compared to Superphénix (6 instead of 8 IHX and 3 instead of 4 pumps) then leads to a main vessel diameter of only 17.2 m.

In the interest of maximising the availability of the plant for power generation operation is possible with one or two secondary circuits isolated because each of the IHXs are individually connected to one of the 6 secondary circuits.

Simplification has been achieved for the internal structures with a redan formed by a single shell directly connected to the diagrid. The main vessel cooling feed has been simplified and a solid reactor roof replaces the fabricated box structure.

Thermal-hydraulics of the Primary System

Management of the thermal-hydraulics of the primary system and particularly the hot pool is most important for assuring the life time endurance of the structures.

Development of the optimum design arrangement has involved scale modelling in water and 2D and 3D computational modelling. This R&D has been shared between the organisations in France, Germany and the UK.

The effectiveness of the mixing of the core outlet jets has been demonstrated successfully and fatigue damage due to jet fluctuations, so called thermal striping, should not be a problem.

To obtain a stable core outlet jet and, at the same time, a quiescent pool surface that does not entrain the argon cover gas, has been a particulary difficult balance to achieve. It needed water models at various scales and computer analyses in order to represent all the controlling parameters.

There has also been important work on the natural circulation behaviour under decay heat removal conditions. The very good potential for a sodium cooled reactor has been demonstrated without severe loadings to the structures.

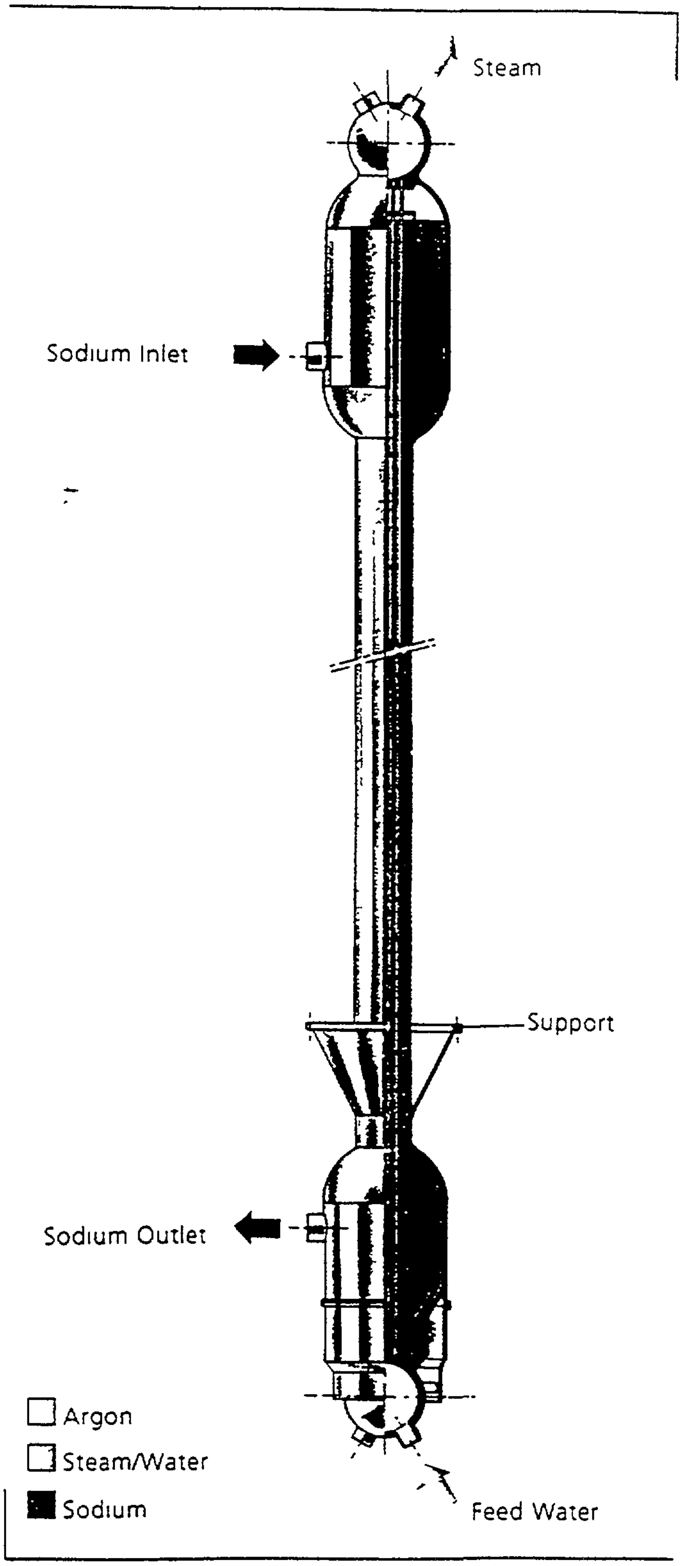


FIG. 12. Steam generator unit.

Main Components

The primary pumps have a top entry, single mixed flow impeller and a flywheel to extend the run-down time on loss of power supply. A decision was taken to opt for a simple design without valves which implies all three pumps must be available for reactor operation.

For the IHX a simple straight tube design is used with the primary sodium on the shell side. Because of the limited head to drive the primary flow, the lower flow resistance on the shell side gives the most compact unit. Extensive analysis has demonstrated that adequate flow and temperature distribution are achieved throughout the operating range.

A new feature is incorporated in the design for sealing the IHX to the redan using piston rings. Similar piston rings are used in PFR but to accommodate thermal movements between the components and the redan requires a novel design. This feature replaces a gas seal and avoids any question of a large gas inventory being held below the sodium surface and offers some reduction in main vessel diameter. The IHX has a valve on the primary side so that the reactor can be operated when necessary when equipment and components are unavailable in the associated secondary circuit and steam plant.

The Steam Generator Units (see Fig.12)are once through straight tube units in high strength ferritic steel. There are no welds in the tubes other than one at each end required to attach the tube to the tubeplates. The design is the most suitable for commercial high quality fabrication and is as simple as could be envisaged but because of the imporatance of its reliability to the availability of the plant the ultimate demonstration will be best achieved with the testing of a prototype unit.

6.4. A Safe Design: Taking Advantage of Favourable Features and Past Experience

The fast reactors possess a number of attractive safety features, such as:

- coolant operating temperatures well below the boiling point, so that dry-out of the core due to coolant evaporation can be excluded;

- very good heat transfer capability and natural circulation behaviour of the sodium coolant;

- low pressure, slightly above atmospheric pressure, in the sodium heat transfer systems;

- use of ductile structural materials allowing for leak detection long before rupture;

- a stable core with negative reactivity feedback to power increase.

The risks arising from the core not being in its most reactive configuration and the use of a combustible coolant were identified at the outset of fast reactor development. Steps have been taken in the design to prevent hazardous consequences, where, for example, design optimisation is possible for the core reactivity coefficients and provision of extremely reliable shutdown systems. The safety features of FRs have been improved continuously. The experience of the various prototypes currently in operation has confirmed the operability and benign characteristics of the fast reactors. This has given grounds for confidence in the future.

An important achievement has been the demonstration of the licensability potential through an independent review of the essential elements of the safety case by a group of prominent safety experts from France, Germany and the UK. This review allowed in-depth discussions on the general approach, the main safety functions and the proposed risk minimisation measures. The outcome was as follows:

- On core melt: adequately low levels of risk are achievable.

- On sodium fire hazards and in-service inspection: significant progress has been made with EFR compared to previous fast reactor designs.

- On decay heat removal: the improvements incorporated in the design are judged satisfactory.

- Generally the approach adopted for EFR results in levels of safety comparable with those of future PWRs.

In conclusion, the final view expressed by the independent experts was that "they were favourably impressed by the progress that has been made with the current EFR design, and they consider that it provides a sound basis from which to proceed to licensing application in each of the participating countries."

6.5. The Approach to Competitive Fast Reactors

More than 40 years of fast reactor research and development, design and operating experience have paved the road towards economic fast reactors, giving confidence that competitiveness can be reached.

There is no unique yardstick to measure competitiveness on a worldwide scale, and not even on the European scene, because of varying industrial, commercial and fuel supply conditions from country to country. On the basis that , in the next century, nuclear energy should provide an important contribution to electric power generation it is neccessaey for fast reactors to compete with advanced thermal reactors, notably the pressurised water-cooled reactors (PWRs).

The demonstration of the economic potential was, therefore, an important objective of EFRdesign studies, leading to thorough investigation of all components of the generating cost, i.e., plant investment, fuel cycle and operation costs.

The economic investigation was conducted as a joint enterprise between EFR Associates and EFRUG, each being responsible for clearly identified areas of work.

Plant Investment Cost

Many lessons have been learnt from the construction experience, the most recent being Superph_nix in France and SNR 300 in Germany. Of course, SNR 300 was beset by political difficulties, from which lessons must also be learnt. However, the construction of both plants provided a wealth of information which leads to the simplification and optimisation of future plant designs.

The design work on EFR has already achieved substantial investment cost reductions for the NSSS compared to Superphénix as is illustrated in Fig.13 which displays the specific weights of steel employed for the main systems of Superph_nix and EFR.

In order to provide a firm basis for the economic assessment of EFR and to confirm the favourable trend deduced from the comparison of economic indicators, the EFR Associates launched a comprehensive cost enquiry campaign which involved experienced component manufacturers in Europe. The manufacturers were asked to provide quotations against specific 'Cost Enquiry Specifications' for a number of components typical of fast reactors.

These specifications required that the manufacturers had to comply with their national standards for the fabrication of nuclear components. This approach avoided cost distortions due to the need for accommodating foreign standards. A total of 21 key components were selected for quotation by manufacturers and corresponding cost enquiries were addressed to 19 manufacturers in 4 countries: Austria, France, Germany and the UK, with two to three quotations per component giving rise to 57 quotations in all.

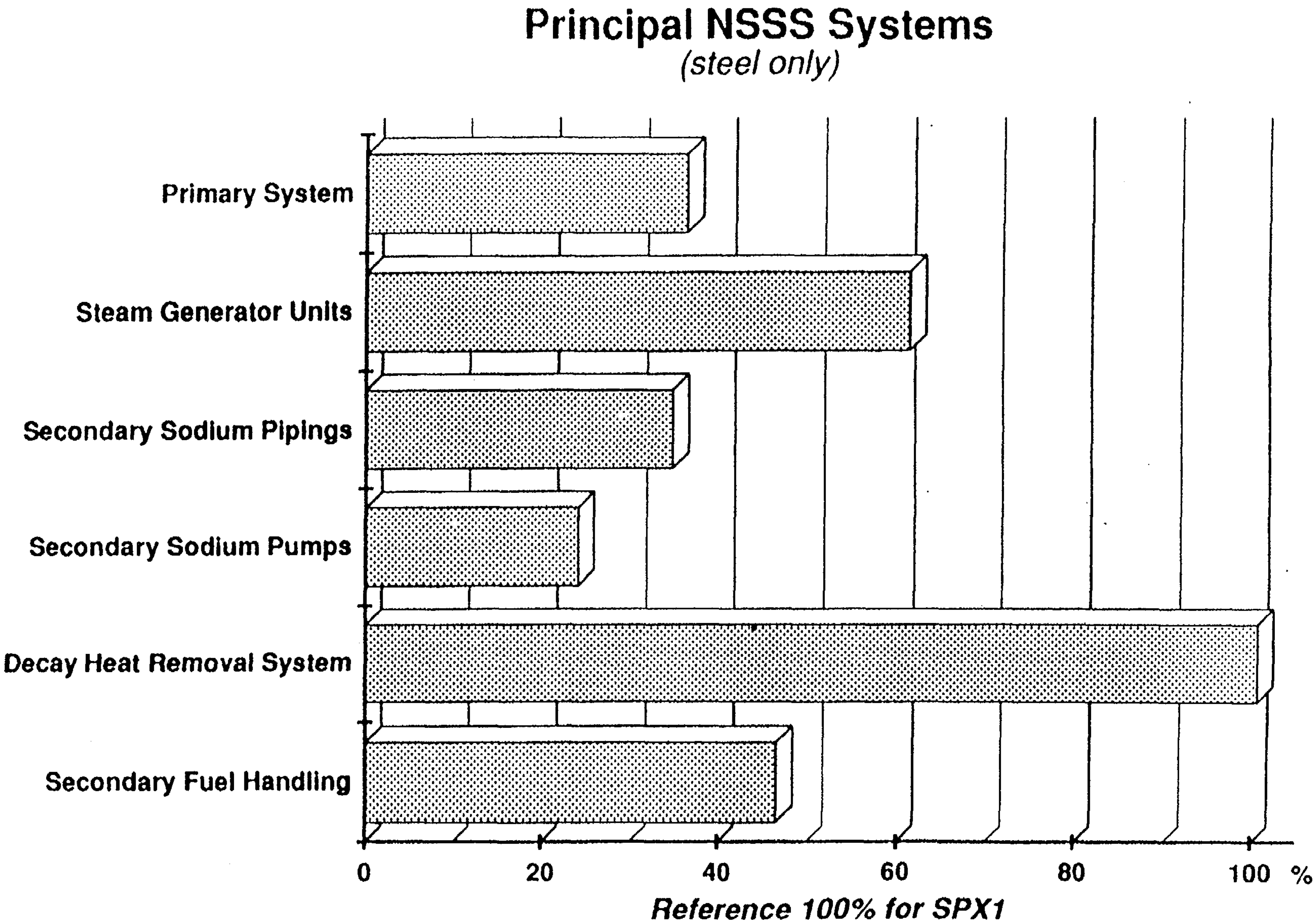


FIG. 13. Comparison of the specific steel weight in te/kW(e).

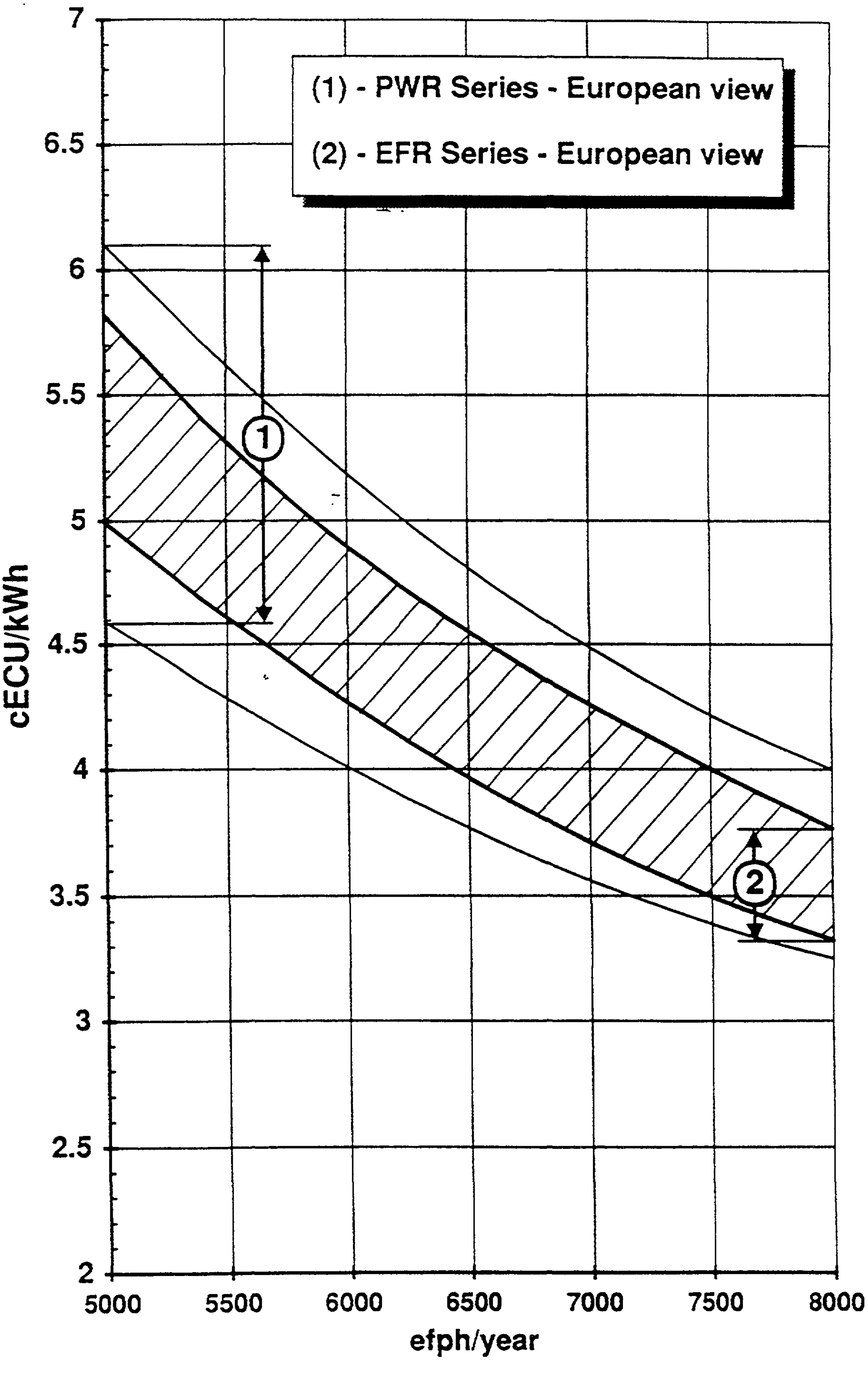


FIG. 14. Generating cost comparison EFR/PWR.

Component costs were obtained for the first-off EFR station and for a series unit in a programme of fast reactors so that cost reduction due to the learning effect and series construction could be fully exploited.

The Nuclear Island cost assessment by the EFR Associates was combined with cost estimates of the Conventional BOP by the utilities of EFRUG to establish the construction cost of the whole station under the conditions prevailing in each country. This provided the main input for computation of generating cost.

Fuel Cycle Costs

Fuel cycle costs are a function of burn-up or mass flow in the fuel cycle. Fuel design for high burn-up thus plays a dominant role in the future of fast reactors and is a major contribution to their competitiveness.

To secure the assessment of EFR fuel cycle costs, EFRUG requested the major companies specialised in nuclear fuel fabrication and reprocessing, BNFL and COGEMA, to provide up-to-date information on fuel service costs. As for the plant construction, the fuel service cost assessment was requested for the first-off station and for a series unit in a programme of fast reactors to establish a fuel cycle operating under similar industrial conditions as for the PWRs.

Initially the fast reactor fuel services will be provided by existing PWR recycled fuel (MOX) fabrication and reprocessing plants. Eventually plants designed for fast reactor fuel services on a commercial scale will improve the fuel cycle performance, leading to a minimum cost expected to be markedly lower than that of thermal reactors.

Generating Cost

The goal of competitiveness for EFR can be considered achieved with the EFR cost falling within the range of PWR costs in Europe (see Fig 14)The range of EFR generating costs being smaller than that for PWR's because the EFR nuclear island is common, only the Conventional BOP is peculiar to each country. It is essential in order to achieve this competitive position for EFR that high plant availabilities, similar to established PWR's, are realised and for the confidence of the utilities this needs to be demonstrated by Superph_nix.

6.6. Status of Design Validation

The specific EFR R&D programme started with definition of the First Consistent Design in 1988. Just as EFR is an evolution of the earlier national designs, the reorientation of the R&D programme was a continuation of the rationalisation process which had been taking place since 1984 when the European collaboration on fast reactors started.

This reorientation was completed by 1990 when all the key R&D requirements for EFR concept validation had been defined and agreed. In 1993 most of the work had been completed by the R&D Organisations with only a few exceptions related mainly to recent design decisions and long term work on material characterisation.

The goal for the core design is a high burnup (20% heavy atoms) and a long residence time in the reactor. Experience from the fuel irradiation in the prototype plants validates the design up to 15 at.% with a statistically significant number of fuel pins, some of them having reached 20 at.%. This gives confidence that 20 at.% is a reasonable target for which validation will be obtained by continuing the experimental irradiation programme.

The compact primary system poses questions for the thermal-hydraulics and structural material behaviour. A very substantial programme has been completed with work sharing between the three countries for tests in sodium and water models of the reactor at various scales. A number of iterations between the design and R&D teams was necessary to identify the optimum design arrangement.

Operating experience with the prototype fast reactors has revealed the critical importance of the steam generators to the availability of the plant for power generation. The key elements of the design of these components have been validated and it remains to demonstrate that the required reliability is achieved through testing of a steam generator prototype. Furthermore, extensive analysis has been completed on the evolution of a leak between sodium and water in the steam generator to establish the soundness of the provisions for accident mastering. R&D programmes are continuing to get confirmation of the material wastage rates and of the feasibility of a highly sensitive leak detection instrumentation to enable an efficient prevention of large sodium/ water reactions.

The potential that the sodium cooled reactor has for the removal of residual core heat by passive means, i.e., without the need for emergency power supplies, has been confirmed by extensive water modelling and analysis of the primary circuit and by tests of a direct reactor cooling loop in sodium. This not only validated the concept, but demonstrated the sodium/air heat exchanger design by large scale prototype testing.

A number of important safety issues have been settled by the validation programme, including performance of the core monitoring instrumentation, natural degasing behaviour of the primary circuit, and, in particular, of the diagrid which feeds the core. Extensive work also has been devoted to sodium fires, and attention given to the experience gained with the Superph_nix restart licensing.

Finally, to give an appreciation of the full extent of the R&D work, there were over 1,000 individual tasks performed at 16 research centres in three countries cooperating for the EFR programme. In addition to these specific tasks, there has been the invaluable experience from operation of the prototype plants which provide a continuous feedback of information for future plant design.

6.7. Conclusion and Outlook

The EFR project at the end of the Concept Validation Phase has matched the ambitious expectations of the customer utilities. This has been achieved as a result of the partnership between the design companies, the R&D organisations and the utilities. The success of the project is a shining example of European cooperation.

At the completion of the Concept Validation phase the economic assessment, involving the key manufacturers in the participating countries, confirmed the potential for a series of EFR to be competitive with the national PWRs.

It was also confirmed that the EFR design and safety features can be confidently expected to be licensable in each of the participating countries. This statement can be made following the positive outcome of the independent safety review completed early in 1993.

The feedback of experience from the operating plants is invaluable in providing assurance that commercial fast reactors can be successfully operated. This has been done and is always an important part of an ongoing activity which has been agreed with EFRUG.

This future programme reflects the evolution of the EFR project regarding timescale and aims at further enhancing the flexibility of the fast reactor for its potential to manage the plutonium stockpile and transmute minor actinides according to the prevailing requirements from the fuel cycle. However, securing the long term fuel supply for electricity generation through breeding remains the main aim of fast reactor development. The objectives of these ongoing activities can be summarised as follows:

1. To establish the flexibility of the fast reactor for managing the back end of the fuel cycle.

2. To evaluate the potential for further improvements which have been identified during the Concept Validation phase, notably regarding in-service inspection and repair and other advanced features which could be introduced within the time that is available for their validation.

3. To assess the design of a demonstration plant of a reduced size from the point of view of demonstrating advanced features of a large plant and as an irradiation facility for the next century.

The next period will seek to maximise the extent and benefit of collaboration:

- within Europe, to include all those parties with a strategic interest, particularly in the full fuel cycle,

- outside Europe, to realise the potential which has been established for extending collaboration to Japan, Russia and the USA.

7 - RAND D.

7.1. Fuel element and core materials

The extended shut-down of PHENIX in 1994 had a major effect on the achievements of the goal for EFR fuel (20 at % BU) and core materials (180 dpa). The programme, as in 1993, was reoriented towards PIE, materials properties assessments and towards CAPRA fuels.

Irradiation progress.

In 1993 no further progress could be achieved in dose/burn-up values. Thanks to the start-up of PHENIX the December 24th, 1994 for the 49th cycle the leading subassembly with the reference cladding (CW 15-15 Ti) reached 160 dpa.

The satisfactory performance of the austenitic material (CW 15 - 15 Ti and 1.4970) at these burn-up levels and the improvements in the specification of the improved CW 15-15 Ti (AIM1 material) give confidence that the defined targets for EFR are achievable.

Material properties.

Austenitic steel materials for clad (CW 15-15 Ti) and wrapper tube (EM10) were qualified respectively for 115 dpa (clad) and 100 dpa (wrapper S-A) on the basis of post-irradiation examinations.

Out-of-pile fabrication and tests were in progress on dispersion-strengthened ferritic stainless-steel.

Fuel modeling.

Some models of the thermomechanical GERMINAL computer code were improved on the basis of PHENIX pin results and some specific CABRI tests. In particular the transient swelling phenomena has been modelized into GERMINAL.

Thermomechanical behaviour of sub-assemblies.

The programme on seismic tests on a bundle is in progress as well as interpretation.

CAPRA fuel.

Characterization of mixed fuel with high Pu content is in good progress. Fuel fabrication in the range (40 - 45 % Pu) is established as well as, providing a few verification, fuel solubility in nitric acide.

Experimental irradiation preparation in SILOE, HFR, PHENIX, SUPER-PHENIX were achieved in 1994. The experiment IFOP1 (45% Pu) in SILOE reactor is completed (35 EFPs) and PIE are in progress.

7.2. - Core physics

7.2.1. - Nuclear data and the "unified code system" ERANOS

The "formulaire" unified code system) ERANOS is a system of coherent tools (cross section data and calculational modules) developped and qualified for general application.

This system bas been developped during more than 20 years, at Cadarache; within the framework of an international collaboration.

Starting from the JEF2 European Data Bank (microscopic cross section data and nuclear constants) the codes NJOY and CALENDF are used to prepare the cross section data libraries used with the ERANOS scheme : (see table 7.1)

ASPILIB 2 (175 energy groups) for shielding studies,

ECCOLIB 2 (33 energy groups) for core physics applications (project route),

EUROLIB 2 (1968 energy groups) for core physics applications (reference route),

ERALIB (1968 energy groups), adjusted and qualified.

These libraries form the basic cross section data for the ERANOS code scheme.

The basic FUNCTIONS of the system satisfy all the requirements of fast reactor neutronic studies : CELL calculations (the code ECCO), CORE and SHIELDING calculations, neutron and gamma PROPAGATION calculations.

In the ERANOS scheme the CELL code ECCO has a resonance self-shiedling solution which is resolved by the sub-group method with an integrated multi-group transport calculation in plane, cylinder, rectangular and hexagonal 2D, and 3D plate geometries.

For CORE and PROPAGATION calculations the ERANOS scheme contains diffusion theory modules (1D, 2D, 3D), and transport theory modules using either the S_N method 1D, 2D, (XY, RZ, Rθ) or the variational nodal method 3D (XYZ, HEX-Z). These modules are capable of providing a direct or adjoint homogeneous or inhomogeneous solution.

Currently under development is a 3D spatial kinetics capability as well as the inclusion of multi-group MONTE CARLO modules.

Together, these modules allow the calculation of the following core physics parameters :

- reactivity distribution,

- fission rate and power maps,

- reactivity coefficients (Doppler, sodium void),

- sensitivity coefficients (exact perturbation),

- reactivity loss and fissile inventory (fuel cycle),

- graphical post-treatment.

The process of QUALIFICATION has the aim of defining for each parameter of interest its method of calculation and its associated uncertainy. The experimental base for this qualification is extensive :

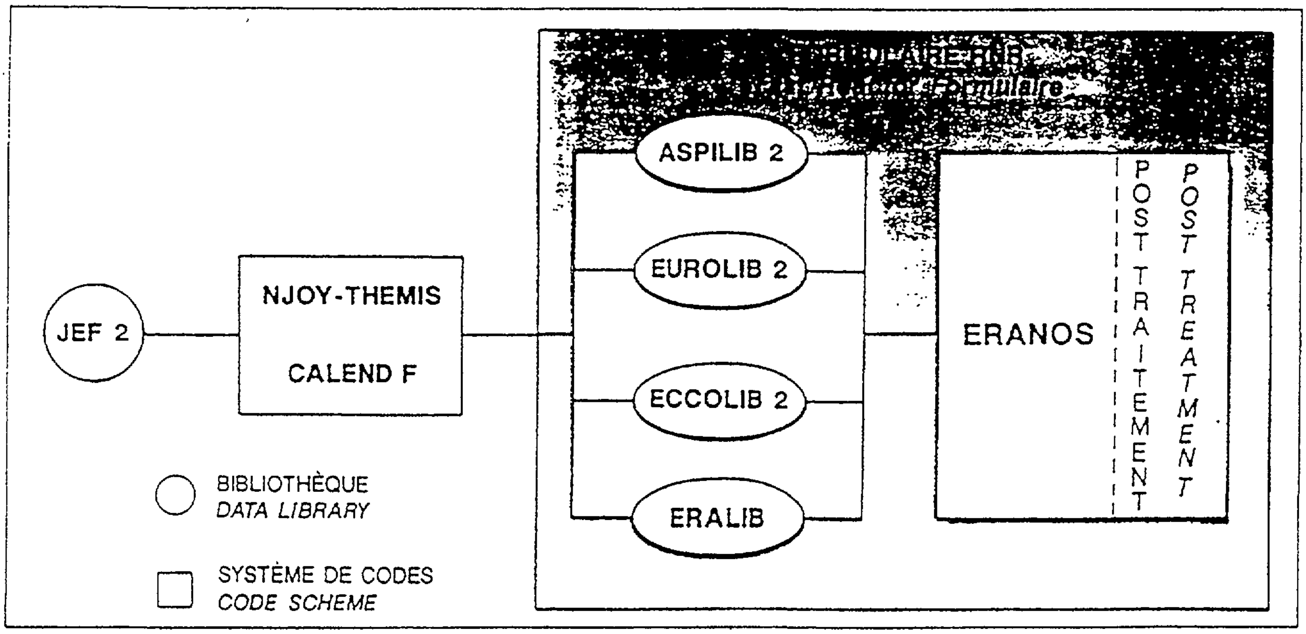
- the RZ, PLUTO, BALZAC, CONRAD experiments performed at the MASURCA experimental facility at Cadarache,

- complementary experiments performed within the European collaboration (ZEBRA, SNEAK, JANUS, etc.),

- previous experiments performed at the PHENIX and SUPER PHENIX reactors.

For each paramater the method of QUALIFICATION allows the separation, in the difference between CALCULATION and EXPERIMENT, of the contributions that arise from deficiencies in basic data and approximations in the calculational method used.

TABLE 7.1. "unificial code system"



This separation also permits :

- the qualification and adjustment of the multigroup data libraries (transition from ECCOLIB to ERALIB). This adjustment, necessary for fast reactor applications has been made possible by the development of an exact statistical model combined with the use of sensitivity calculations,

- the qualification and improvement of the calculational algorithms and the validation of the physical approximations assumed, depending on the calculational route chosen (project route or reference route).

En 1994 studies were devoted to the DOPPLER effect in the CAPRA core, it was shown that iron has a significant but delayed effect.

ERANOS 1.1 and ECCO 5 versions were released.

Improvement of the cross section date libraries was in progress by continuing qualification on integral experiments.

7.2.2. - BERENICE programme

The BERENICE programme (Beta Effective Reactor Experiment for a New International Collaboration Evaluation) has been carried out at the MASURCA critical facility (CE CADARACHE) with the participation of an european team (UK (AEA-T), Italy (ENEA) and France (CEA)) and teams from Japan, (JAERI), USA (LANL) and Russia (IPPE).

The goal of this programme was to reduce uncertaincies as to the effective delayed neutron fraction, βeff, from 10 % ((2 σ) to 5 % (2 σ) in order to obtain better prediction of the reactivity scale.

Initially, three configurations were proposed in order to separate the contribution of the main isotopes :U^{235}, U^{238} and Pu^{239} ; as shown in the following table, constitution of the 3 differents cores :

FUEL	Cell	Height (cm)	Radius (cm)	F8/F9	Enrich. %
U	"R2"	61.6	49.5	0.0427	30
Pu	"ZONA2"	61.6	52.5	0.0423	25
Pu	COMPACT	61.6	79.6	0.0296	11

Calculated values in PCM (10e-5) with CARNAVAL IV + TUTTLE data :

CORE	REGION	U_{235}	U_{238}	Pu_{239}	Pu_{240}	Pu_{241}	Pu_{242}	Total
"R2"	Core	570.2	140.6					710.8
	Blanket	4.1	28.9					33
								743.8
"ZONA2"	Core	4.4	130.8	151.2	13.7	11.9	0.8	312.8
	Blanket	3.9	28.6	/	/	/	/	32.5
								345.3
COMPACT	Core	18.8	226.9	122	8.7	9.4	0.5	386.3
	Blanket	2.3	14.8					17.1
								403.4

The study of the first configuration "R2" cell was completed in 1993 and the study of the second configuration "ZONA2" cell was completed in March 1994.

Two particular techniques were used to measure the βeff :

- the Californium source pseudo-reactivity method,
- the pile-noise method.

Results and calculated values of the two first configurations are compared in the following tables (values publicated at the "Topical Meeting on Advances in Reactor Physics, KNOXVILLE (USA) April 1994" were slightly corrected due to a revision of the power).

Core	Experimental values (pm)	
	Cf252 source technic	Noise Method
R2 (U^5 enriched : 30%)	724 ± 23 (1 σ)	708 ± 13 (1 σ)
ZONA 2	356 ± 11 (1 σ)	338 + 6. (1 σ)

It is shown that experimental methods allow the reduction of uncertainlies according to the goal.

Cellule Combustible ZONA2 (PuO2/UO2/Na)

Na	PuO2	Na	PuO2
PuO2	Na	PuO2	Na
Na	PuO2	Na	PuO2
PuO2	Na	PuO2	Na

Na = Réglette Sodium

PuO2 = Réglette Plutonium Uranium Oxyde (E=25%)

Cellule Couverture UO2 Sodium (UO2/Na)

Na	UO2	Na	UO2
UO2	Na	UO2	Na
Na	UO2	Na	UO2
UO2	Na	UO2	Na

Na = Réglette Sodium

UO2 = Réglette Uranium Oxydc

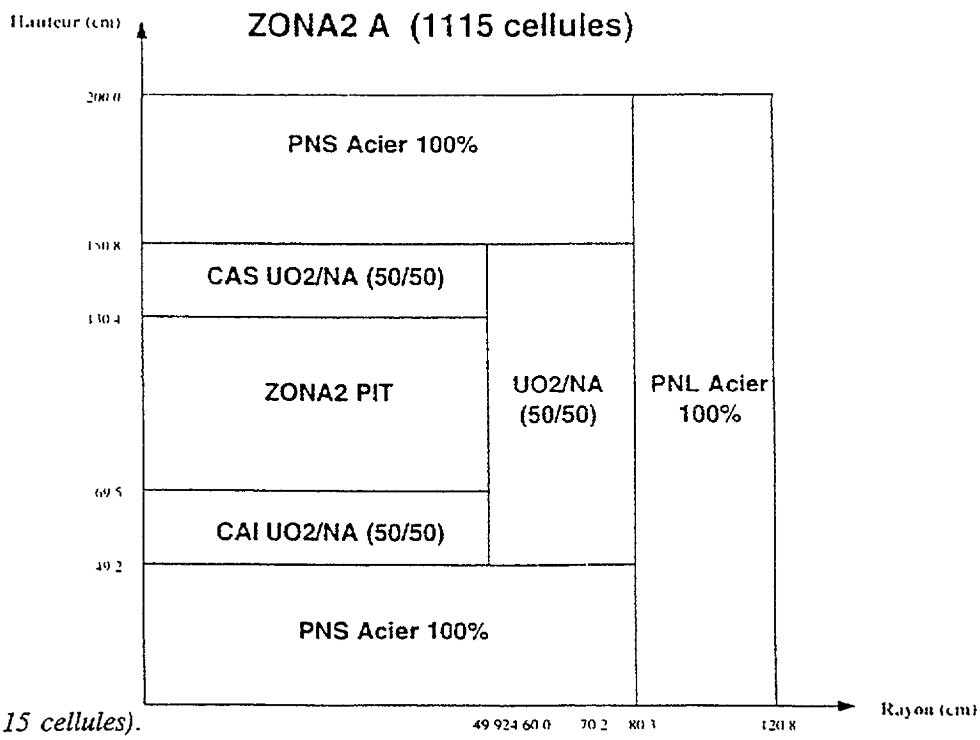


FIG.15. ZONA2 A (1115 cellules).

In order to reduce uncertaincies regarding U^{235}, U^{238} and Pu^{239}, the third configuration should have been studied. In fact this will be done within the framework of the collaboration with Japan (JAERI) where 3 additional configurations are to be studied in the FCA facility.

After completion of the Japanese programme, the third configuration COMPACT could be studied in MASURCA.

7.2.3. - The CIRANO programme

The CIRANO programme is one of the different programmes carried out in order to demonstrate the feasibility of a fast reactor in burning as much plutonium as possible.

In order to reach this goal the U^{238} content must be as low as possible consequently with regard to breeder cores , the following is required :

- suppression of fertile blankets,

- increase plutonium content,

- substitution of inert material fot he U^{238}.

The CIRANO programme is conducted at the MASURCA critical facility. The first topic presently being studied.

Only a few experiments were performed without fertile blankets and stainless steel reflectors and it seems that the discrepancies between expriments and calculations increase as the thickness of the blanket decreases.

Also the first set of studies consists in gradually removing blankets from the ZONA 2A (BERENICE Core) as is shown is the following shedule

April 1994	ZONA2 A	reference core (Fig. 15)
September 1994	ZONA2 A3	radial blanket removed (Fig. 16)
February 1995	ZONA2 B	core without blanke (Fig. 17)
July 1995	ZONA2 B	core without blanket and with an internal storage
November 1995	Set of substitutions in the centre of ZONA 2B for parametic studies.	

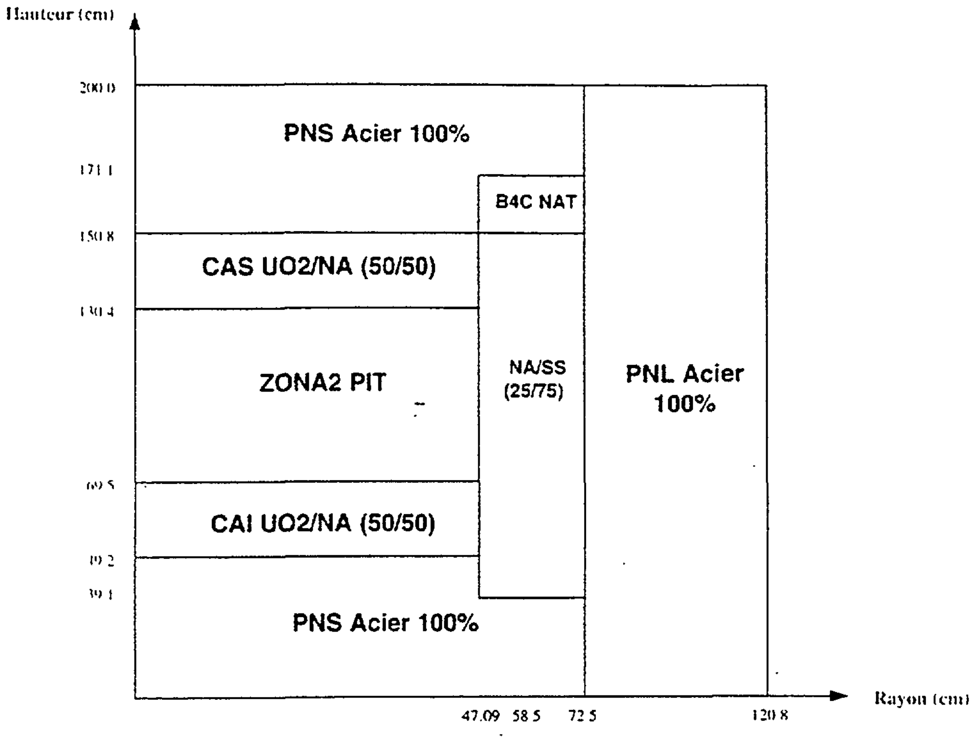


FIG.16. ZONA2 A3 (992 cellules).

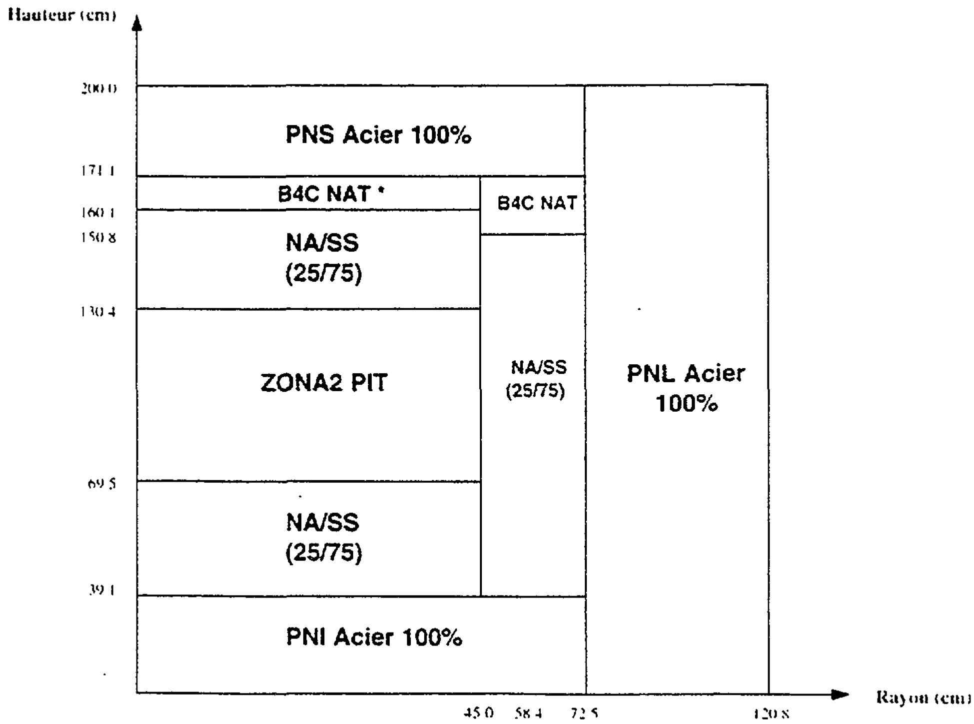


FIG.17. ZONA2 B (906 cellules).

7.3. - Safety

7.3.1. - Reactor dynamics in normal and design basis accident conditions

The computer code DYN 2B for plant dynamics has been fully validated with the SUPER-PHENIX start-up tests and has been documented. The OASIS computer code, a flexible generic tool for a wide range of reactor systems, has been qualified by cross comparaison with a first set of DYN 2B calculated transients. Qualification on a second set is almost complete.

7.3.2. - Core surveillance

Core surveillance in a continuous R and D task.

By separating the different components of the temperature signals, the numerical filtering technics considerably improved individual monitoring on the subassembly for early detection of incident if not accident. Possible slow evolution in the core can be followed in a more satisfactory manner while continuing detects any rapide event using treatment limited to operation which might be carried out on line. A demonstration system the been developped, the ALPES system.This system will be tested on line using the temperature signals directly measured on SPX (a paper will be published at the SMORN-VII Conference in AVIGNON (FRANCE), June 1995).

7.3.3. - SCARABEE - N Programme

A synthesis report on the SCARABEE - N programme has been issued. The SCARABEE tests were realized by CEA-IPSN, but with the cooperation of several french and foreign partners, such as EDF and CEA-DRN/France, AEA/UK, KFK/Germany, PNC/Japan and JRC/EU.

The scarabee-N in-pile tests were performed between 1983 and 1990 with the main objective to study the consequences of a hypothetical total instantaneous blockage at the inlet of a LMR sub-assembly at full power.

After fourteen tests, mainly with fresh fuel, this accident scenario may fairly well be described.It has been shown that no violent, energetic fuel-coolant interactions take place, that almost no fuel is ejected out of the fissile zone and that the melt penetration into the neighbouring sub-assemblies proceeds fast.

Progress has been made on the understanding of the behaviour of mixed boiling pools and the hexcan melt-through thresholds could be established.

Codes have been developed (PHYSURA-GRAPPE, SURFASS), others have used the SCARABEE-N tests as contribution to the validation (SIMMER II, SABRE) and much interesting information for other accident situations was made available, especially for the transition phase.

7.3.4. - Fuel pin safety tests in CABRI

Extensive analysis of the CABRI-2 tests has been achieved and a synthesis report of the CABRI-2 programme entittled : "Results and Achievements" was issued recently. This international programme was carried out with the participation of CEA/IPSN (France), KfK (Germany), PNC (Japan) and AEA-T (UK).

At the present time, the CABRI-FAST international programme is still continuing. In this programme the behaviour of annular fuel cladded with the improved (low swelling) clad material 15-15 Tiε (CW 15 - 15 Ti) is studied under slow ramp and TUCOP conditions.

The test matrix and characteristics of the pins are shown in the following tables.

	1992	1993	1994	1995
CABRI - FAST TEST MATRIX	BCF1 (12/92)	PF2 (3/93) PF1 (12/93)	LT2 (3/94) EFM1 (5/94)	PFX MF2 LT1
EUROPEAN TESTS	JOG2	JOG1		

TEST	OBJECTIVES	PIN
Slow ramp rate tests	-	
PF1	Pre-failure fuel behaviour (1% Pn/s)	SCARABIX
MF2	Margin to failure (1% Pn/s)	SCARABIX
BCF1	Beyond clad failure Ramp effect (3% Pn/s)	VIGGEN - 4
Initiation phase tests (CDA)		
PF2	Pre-failure behaviour Fuel squirting	QUASAR
LT2	TOP. Fuel squirting	QUASAR
LT1	LOF-TOP. Fuel squirting	QUASAR
EFM1	Extended fuel motion	SCARABIX

NAME	TYPE	CLAD	D_{out} mm	BU at%
VIGGEN-4	PX homogeneous solid (no UAB)	15-15 Ti	6.55	11.7
QUASAR	PX homogeneous annular (no UAB)	15-15 Ti	6.55	11.8
SCARABIX	SPX homogeneous annular (UAB)	15-15 Ti	8.5	6.4

Two tests were carried out is 1994 : LT2 and EFM1.

The goal of the LT2 test was to study internal molten fuel motion (fuel squirting) in TOP conditions ; no rupture was observed and the maximum energy deposit was 1.26 KJ/g.

The goal of EFM1 was to study molten fuel motion and freezing in the voided coolant channel in TOP conditions and after clad rupture.

7.3.5. - Hypothetical Core Descriptive Accidents (HCDA)

7.3.5.1. - Initiative Phase

The policy agreed upon by the European R and D organizations on developing the FRAX ,PHYSURAC, SAS 4A initiation phase codes was as follows.

Work on the development of the FRAX code at AEA-T was concluded in 1993. The final FRAX 5D version was issued in March 1993.

Work on the development of the PHYSURAC code at the CEA has also been completed. Future work will be limited to the maintenance of this code.

The other code, SAS 4A has been further developed, fully taking into account the data from the CABRI 1, CABRI 2, CABRI FAST tests, and other relevant tests, together with advances in the theoretical modelling of phenomena.

This development is being carried out in close collaboration with KfK, the CEA the PNC and still with a small participation of ANL and AEA-T. There are agreements existing between the Argonne National Laboratory with KfK, AEA, CEA-IPSN and PNC respectively. The pratical collaboration in ensured by a SAS 4A users' group which meets once or twice a year. A "code manager" is in charge of maintenance, the Quality Assurance and the introduction of improvements for the benefit of the users. The "code manager" up to the end of 1994 was KfK. The next manager is CEA-IPSN.

7.3.5.2. - Transition Phase

Following a better understanding of the SIMMER II and Advanced Fluid Dynamics Model (AFDM) codes, the European have decided to join the PNC (Japan) to developed SIMMER III. PNC is the code developper.

The main task of the Europeans is now to participate in the qualification of the code and neutronic module improvement.

7.3.6. - Source term and radiological safety analysis

In Europe a considerable amount of effort has been exerted to derive data in the release of fission products and fuel from the molten core to the sodium, and transport to the cover gas, and to develop codes to model the various phenomena. Future R and D work should focus on the evaluation of the release of fuel and fission products in the cover gas for different accident scenarios, taking into account the above data and codes.

The transport of the fuel and fission products from the cover gas to the various cells of the reactor building and to the environment is calculated with the CONTAIN-LMR code. In 1994 in CEA the work was limited to maintenance and applications of CONTAIN-LMR.

7.3.7. - Sodium fires

Experimental work et CEA-CADARACHE (IGNA-programme) and KfK (FAUNA tests) on combined pool-spray sodium fire test with sodium leaks up to 230 kg/s was completed in 1993. In 1994 the results were analyzed and used with previous tests for qualification of the FEUMIX 3 code.

In order to have data to extrapolate the sodium experimental results for large leak rates (a few tons/s) the AIRBUS programme is now in progress. The goal of this programme is to measure the Sih coefficient (interfacial area multiplied by a heat transfer coefficient) which is used in the modelling of the FEUMIX 3 code. The AIRBUS programme is being carried out in water and air and in similarity with sodium conditions.

7.4. - Technology

7.4.1. Information about the March 31 st, 1994. Accident in Cadarache.

On March 31st, 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 occured. One member of the CEA staff was killed and four people were wounded.

A commission of inquiry chaired by the CEA General Inspector for Nuclear Safety was set up by the "Administrateur Général" of the CEA the day following the accident. This Commission delivered its first conclusions in July 1994.

The sodium present in the tank in which the accident occurred comes from the primary cooling circuit of Rapsodie. After the final shutdown of the reactor, in 1982, all 37 metric tons of primary sodium were drained in 1985, then filtered to be purified from most of their radioactive contamination ; the 137 cesium activity, for instance, was reduced to about one hundredth of its original value.

After these operations, this sodium was stored for eight years in the tank, out of which it was syphoned in the second half of 1993 to be transformed into sodium hydroxide at the Desora facility. This facility, located inside the containment building of the reactor, had been especially designed for that purpose. When the tank was drained, there remained at the bottom, a residual sodium quantity.

Before dismantling, the tank had to be cleaned in order to remove this residual sodium. The process selected to perform this clean up operation, already implemented several times and notably in 1986 in a tank of similar geometry, consisted of progressively introducing in the tank a heavy alcohol, called ethyl-carbitol (CH_3-CH_2- O - $(CH_2)_2$ - O -(CH_2) - OH), while monitoring the reaction through temperature, pressure, hydrogen and oxygen measurements.

In its first conclusion the inquiry commission considered that the major cause of the accident was due to the formation of an heterogeneous physical-chemical environment, complex and multiphasic made up of three basic components (alcohol, alcolate, sodium). This environment developped after a cooling during several days due to a halt of alcohol injection, followed by an external electrical heating complemented by the local heat produced by the reaction themselves after resuming alcohol injection during the next last two days before the accident. This environment turned out to be particularly favorable 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 March 31st, the phenomenum 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.

The state of available knowledge did not allow the loss of control over such chemical reactions to be predicted or anticipated ; chemical reactions whose exact nature had not yet been determined. Consequently the Commission proposed a programme of detailed local investigations and a programme of complementary studies and analyses which involved basic knowledge of chemical reactions and chemical analyses of the reaction residues and metallurgical, mechanical and thermal analyses and analyses linked to the human factor. Also the Commision proposed a programme of global chemical experiments in order to understand the integral phenomena.

Laboratories of the CEA, the chemical industry, the university and the CNRS were involved in carrying out this programme.

Consequently and in conformance with the commission of inquiry recommandantions :

- a project manager was appointed to carry out complementary studies,

- washing of sodium pools using alcohol will no longer be authorized as long as the studies and complementary appraisals have not been completed - an increased vigilance will be included in the design of the capacities containing dangerous products in order to facilitate future dismantling. Further vigilance will be required regarding the use of solvents and derived products, and regarding the presence after the final shutdown of a facility of a significant team having an excellent hnowledge of this facility.

7.4.2. Sodium technology.

Studies regarding the set-up of one oxigen-meter and one tritium-meter in an auxiliary circuit with PHENIX primary sodium are in progress.

Experimental and theoritical studies regarding optimisation of the sulfophosphoric decontamination process, used for 316 SS, especially for IHXs were completed.

The aim was to lower the phosphoric acid content ; this acid prevents intergranular corrosion but leads to undesirable phosphate in the waste.

Regarding sodium leak detection, qualification of Sandwich type detectors (metallic foils inside tow silicoaluminate layers) is completed.

A computer model is being developed to predict the movement of sodium from a leak beneath the pipe insulation, to estimate likely corrosion and to predict the efficiency of detector devices.

7.4.3. In-service inspection and repair.

. Main vessel inspection.

Most of the activity was devoted to improving the MIR system. A work of modelling of ultrasonic waves in austenitic steel is in progress ; special attention is given to the inspection of the "triple point". (Where the core support structure rests on the main vessel) as well as inspection of the lower part of this support structure.

Complying with the current specification leads to increased requirements for ultrasonic transducers, emphasising the potential interest of array-transducers.

. In-gas telemetry.

The model describing the deviation of a light beam above a sodium pool due to thermal heterogeneities was completed and is now optimized. The problem of aerosols remains to be resolved.

. Structural monitoring in sodium.

After the extensive study accomplished in 1993 additional studies were undertaken :
- the possibility not to use specific targets, keeping a satisfying precision
- the signal processing
- possibility of visualizing ultrasonics is being explored.

. Above sodium repair.

The MIRSA facility ("maquette pour interventions et réparations sous argon") which is a facility for Intervention and Repair in Argon is ready for operating. In 1995 the programme will consit in welding tests (samples wet with sodium or clean).

7.4.4. - Steam generator.

. Water sodium reaction.

A new model to calculate the "wastage" was introduced in the PROPANA computer code. The hydrogen detection model was also improved.

A series of bursting tests on Alloy 800 was completed in 1994 and will be used for qualification of the MECTUB code.

. ATLAS monotubular rig (Fig.18)

The ATLAS mock-up test programme, devoted to detailed qualification of heat transfer, pressure loss and dry-out correlations, is scheduled to start in spring 1995.

Test section manufacture was completed and installation with connection to the steam-water and sodium supply systems of the SET (Station Essais Technologiques : Technological Test Section) facility was carried out in 1994. Cleaning of the sodium took longer than expected.

. Leak detection in steam generator units (SGUs).

SG leak detection experiments were performed by AEA-T teams after the PFR shut down. A CEA team followed these very interesting experiments ; analyses of the tests are still being carried out.

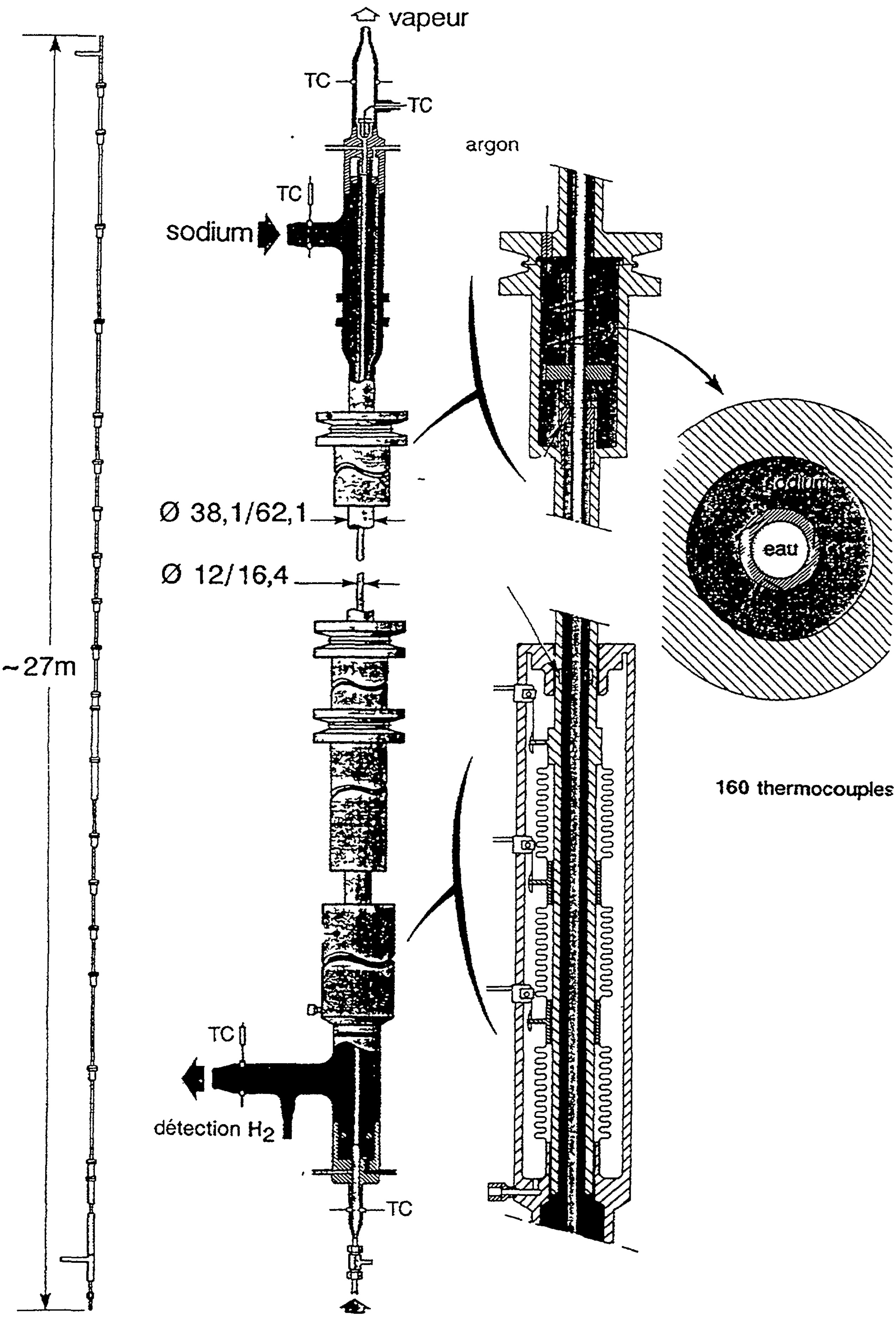

vapeur
TC
TC
argon
TC
sodium
Ø 38,1/62,1
Ø 12/16,4
~27m
sodium
eau
160 thermocouples
TC
détection H_2
TC

FIG 18

7.5. Materials and mechanics

7.5.1. - Materials.

7.5.1.1. Steam generators - Alloy 800.

Alloy 800 is used for steam generator tubing in SUPER PHENIX. Modified 9 Cr 1 Mo steel has been selected for steam generator tubing in EFR, but Alloy 800 remains the back-up material.

A considerable amount of mechanical properties design data on Alloy 800 has already been produced for SPX1, but certain additional work, especially long creep tests were necessary for confirmation. These long term creep/stress rupture tests include the effects of titanium and aluminium additions and welding evaluations. Work has been carried out mainly on pre-pressurised tubes tested at temperature 525° and 550°C.

Corrosion studies were carried out on tubes with and without welds at slow strain rates ($\sim 10^{-7}$ - 10^{-8} s^{-1}) in caustic solutions (0.1%, 1%, 10% Na OH solution) and in water plus hydrogen. No cracks were observed except in the welds for high caustic solution. Corrosion studies were also conducted on materials in the as manufactured state ageing for several hundred of hours at 560-570°C. The tests were carried out at constant loads in caustic solutions.

7.5.1.2. EFR straight tube steam generator material (Mod 9 Cr 1 Mo steel).

The Mod 9 Cr 1 Mo steel has been developped recently and consequently the data available, particularly for time dependent properties were relatively scarce. This was particularly true for the thick section tube plates of the once-through steam generator design of EFR, where through thickness properties an difficult to check. In 1994, tests were completed on thermomechanical treatment (various cooling rates) and chemical compositions on the properties of thick Mod 9 Cr 1 Mo plates. Also tests were carried out to determine resilience and fracture toughness of ageing material. All these data are being synthetized.

7.5.1.3. Low carbon austenitic steel.

A special effort is now underway to understand the long term cracks observed on weld and heat affected zone (HAZ) on some A321 PHENIX components.

7.5.1.4. Carbon steels.

Resilience and fracture toughness were carried out on A42 and A48. Tests in sodium on specific mock-ups were completed on 15D3 and in progress for A42 and A48. The purpose of this programme is to accomplish work on the defects observed in the fuel storage and transfer drum of SPX, built in 15D3, and to check that similar defects might not appear with A42, A48 material.

7.5.2. Mechanics-structural integrity.

7.5.2.1. Thermal buckling.

In LMFBR, thermal buckling could be induced by the large thermal gradients which appear at the free level of sodium. These thermal gradients can be fixed spatially by constructive devices or moving under the liquid thermal expansion.

For fixed level of sodium the results are in a mature state.

For moving sodium level the interpretation of the **VINIL** tests (ratcheting at the free level of sodium of large cylindrical shells, with a ratio of radius versus thickness between 100 and 1000) yielded information and trends for the improvement of constitutive models. Moreover ECCO tests (ratcheting and buckling of large cylindrical shells, tests in water) are in progress. A first campaing of tests was carried out on the study of the coupling between buckling and ratcheting mechanisms when the thermal gradient is sufficient and the shell is sensitive to geometric instability (thin and slender structure).

7.5.2.2. Thermal striping.

Status of the art on this subject was given during the Technical Committee of the IWGFR in Aix-en-Provence (November 1994).

7.5.3. Dynamics mechanics structural response due to sodium/water reaction.

The PLEXUS computer code is used to calculate the structural repsonse due to sodium/water reaction. In its current state, PLEXUS analysis can be 1D, 2D, or 3D and takes into account :

- fluid-structure interaction in 2D or 3D,
- radial plasticity and support reactions in 1D calculations,
- 3D/1D connections,
- reaction zone, coupled with the water circuit,
- circuit singularities,
- gravity effects,
- initial flow rate in the pipework.

STATUS OF FAST REACTOR RESEARCH IN GERMANY

G. HEUSENER
Forschungszentrum Karlsruhe Technik und Umwelt,
Karlsruhe, Germany

Abstract

The fast reactor activities of FZK are part of the Nuclear Safety Research Project. The R&D program of this project has been restructured in accordance with the demands of the Federal Government. The key issues and tasks of the program concern LWR and FBR safety and transmutation of minor actinides.

In February 1995, a contract was signed by German and French utilities and reactor manufacturers including Nuclear Power International, to enter a Basic-Design Phase for the European Pressurized Water Reactor, EPR. An accord was also reached between the French DSIN and the German Federal Ministry BMU on basic safety requirements for future nuclear power stations.

1. Energy Policy in Germany

Whereas electricity generation in eastern Germany is still essentially based on lignite (91%), nuclear energy contributes 34% to western German electricity generation. There are 21 nuclear power station units with a total power of about 23 GW in western Germany, and transmission lines are under construction between western and eastern Germany.

Some recent important decisions and developments in German energy policy are listed below:

- The German Constitutional Court ruled in 1994 that the indigenous hard coal subsidies paid by power consumers are unconstitutional. So far these handouts amounted to about 7 billion DM per year. The Federal Government is now seeking a different solution to keep German coal mines alive.

- A new energy bill has been passed by both houses of the Federal Parliament. It enhances the importance of „direct" final disposal of nuclear waste, i.e. disposal without prior reprocessing - a path preferred by the utilities for saving costs.

- The bill requires that even accidents that are excluded for all practical purposes, should not necessitate any incisive external emergency measures in case these accidents happen. In addition, the bill raises nuclear accident liability insurance levels, and modifies the role of the Federal Government in promoting nuclear energy.

- The so called energy consensus discussions between various German parties were resumed in order to achieve a basic agreement between the Federal Government and the opposition on the basic long-term features of the energy policy in general and nuclear plants in particular. Two meetings were held. Common points of view were reached with respect to the importance of renewables and energy saving.

- However, the judgement on Nuclear Energy remained controversial. As a consequence of this, the talks presently are interrupted and will not be continued prior to June/July.

- MOX fabrication in the Siemens nuclear fuel plant near Hanau will not be resumed.
  This decision was taken in view of the strong resistance of the State Government of
  Hessen against any nuclear activities.

- The dismantling of the SNR 300 plant is continuing. The fuel elements of the reactor have not yet been sold.

2. R&D Activities

Several research projects related to innovative systems are being continued at FZK. Some of them have a close connection with former fast reactor R&D and are performed in international cooperation.

2.1 Decay heat removal by natural and forced convection in the primary system

The R&D programme in this field can be subdivided in the following parts:

- Water tests in tank models (e.g. AQUARIUS, RAMONA, GODOM 2, NEPTUN)
- Water tests in the complete DHR chain (KIWA)
- Sodium experiments in the sodium/air heat exchanger circuit (ILONA)
- Code development (e.g. FLUTAN, TRIO, ARTEC).

In the frame of the experimental investigations a broad range of different operating and design parameters were taken into account, e.g. steady state and transient, core power, asymmetric DHR by DHX, type of DHX, reduced fluid level, operation of air dampers and environmental influences.

The experimental work was accompanied by an extensive calculation effort with the codes mentioned above. The comparison of computed and experimental results has provided a successful demonstration of the potential of 3D and 2D modelling. Qualitatively good results were obtained, quantitative results are depending on the modelling of the codes. The R&D activities in this field are terminated.

On the basis of all these experimental and theoretical results it can be concluded that for a FR of EFR-type the decay heat can be removed by natural convection only. The decay heat removal is functioning even under extreme conditions, e.g. delayed operation of DHX, loss of several DHR loops, reduced fluid level, inadverted opening of the air outlet dampers.

Concerning the objective "passive safety" the DHR by pure natural convection is an essential feature to enhance the reliability of DHR.

2.2 HCDA code development and safety analyses

Within the SIMMER-III cooperation between PNC and the European partners FZK, the Forschungszentrum Karlsruhe (FZK) is contributing both in the code development and the code assessment. The main contributions to these items are:

- in the code development field:
  * formulation and improvement of equation of state
  * improvement and assessment of numerical scheme
  * introduction of advanced neutron flux calculation (TWODANT type) into the neutronics module

- in the field of code assessment:
  * sloshing of liquids and compaction processes
  * shockwaves in single- and two-phase media
  * freezing simulation in tubes with AL_2O_3-thermite (THEFIS experiments)
  * code comparison for impact calculations of liquid slugs on rigid surfaces (SIMMER-III versus PLEXUS)
  * analysis of the simulation of flow instabilities

On basis of the experimental information provided by different CAPRI programmes (see 2.3) more advanced versions of the SAS 4A code to study the initiation phase of whole core accidents were developed. This was jointly done by the partners IPSN, PNC and FZK and the last version of the code SAS 4A, REF94.R1, has been released to the contributing partners.

Analyses of an unprotected loss-of-flow for the EFR-1m- consistent core design have been repeated using the improved SAS4A system. Results have shown that consequences of the initiation phase of the accident are significantly reduced compared to those calculated in the previous analyses using FRAX-5D, PHYSURAC and previous SAS4A versions. Differences are mainly due to improvements in the simulation of the fuel pin behaviour during steady state power operation up to high burnup levels and in particular in the simulation of the post-failure materials relocation.

In addition to these project related activities comparative calculations have been initiated for a severe accident (unprotected loss-of-flow) in the BN 800 reactor with a near zero void core design. This exercise is performed under the leadership of the IAEA and is supported by the CEC. In a first consultancy meeting in Vienna on 5/6th December 1994 aspects of the proposed exercise were clarified and agreed between participating partners. These are the Russian Federation (IPPE), Japan (PNC, Hitachi), India and West European countries as Germany (FZK), France (IPSN), UK (AEA Technology) and Italy (ENEA). This comparative exercise has the following five stages:

I Case set-up (input data specifications)
II Steady state fuel pin characterisation
III Pre-boiling transient
IV Boiling transient
V Post-failure transient.

Currently work concentrates on stage I to be finalised together with first results of stage II up to mid 1995.

2.3 Fuel pin safety tests in CABRI

Review of contract conditions agreed upon between senior and junior partners contributing to the CABRI-1 experimental programme led to the conclusion that results of the CABRI-1 programme as documented in the respective synthesis report can be made available now to interested partners of the international research and development community on request to the CABRI project management at the CEN Cadarache, France. Analyses of results of the CABRI-2 programme have been finished and a synthesis report of the CABRI-2 programme had been issued end of 1994 by the involved partners. These were IPSN, FZK, PNC and UKAEA.

Results of the experimental programme CABRI-FAST are becoming available continuously. They contribute successfully to a better understanding of the steady state and transient behaviour of fuel pin designs with hollow fuel pellets. The behaviour of fuel pins of different designs and with medium to light target burnups was investigated. An experimental data base for the analysis of the prefailure in-pin fuel relocation is now available as well as experimental information on the behaviour of a hollow pellet fuel pin design when a superprompt critical power transient is imposed several seconds after boiling onset. The respective information provides an extremely valuable data base for the continuous improvement of models simulating the preirradiation behaviour of hollow pellet fuel pin designs, the transient fuel pin mechanics behaviour and the prefailure as well as pont failure materials relocation. The respective model development contributes mainly to a progressive improvement of the SAS 4A code.

2.4 Transient fuel pin performance tests

A great variety of fast breeder fuel pins have been tested under operational transient conditions in sodium-cooled capsules in the High Flux Reactor (HFR) in Petten/Netherlands. The test field included power ramping, overpower, power-to-melt and temperature transient experiments and the irradiation was terminated end of 1993. In 1994 post-irradiation examinations and evaluation work were made leading to interesting results:

- In the test KAKADU 30 an overenriched fuel pellet within a normal fuel stack (power peaking factor 2.25) was irradiated for a few days; the overpower of about 900 W/cm in the "hot spot" fuel pellet did not cause overheating.
- In the test POTOM-4 three fuel pins containing 15, 20 and 30 % of Pu were preirradiated under nominal power conditions of about 500 W/cm and then by shuffling to a higher neutron position attained about 800 W/cm. In the neutrographs of all 3 pins fuel centre melting is clearly to be seen.
- In one specially designed KAKADU capsule 2 MOX fuel pins (preirradiated to 9 % burnup in PHENIX) were irradiated at 450 W/cm nominal conditions during a cycle of 26 days and then brought to the maximum power of 720 W/cm, i.e. a power ramp of 160 % of nominal power with a power rise of 1 % per second. Despite the high burnup of 9 at% and a neutron dose of about 70 dpaNRT the fuel pins remained intact, showing a slight diameter increase up to 50 microns only.

2.5 Transmutation of Minor Actinides

The activities of the project have been concentrated on two fields:
3.4.1 on the participation in the European CAPRA programme and
3.4.2 on chemical studies for the separation of Actinides from Lanthanides by solvent extraction.

2.5.1 CAPRA project activities

The common cooperation in the frame of the CAPRA project required a concentration on specifically selected topics in the various organisations. Based on that we examined in our neutron-physical work the specific influence on Np-additions to the oxide reference core of the CAPRA burner reactor type. In these studies defined quantities of Pu were replaced by an equivalent amount of Np in the mixed oxide fuel and the resulting change of the most important safety-related properties of the core (Doppler, Na-void, K_{eff}, etc.) were calculated. The results clearly show that a homogeneous addition of Np in the range between 5-10 at% (replacing the same amount of Pu) has a negative effect on Doppler and Na-void and in order to reach comparable values for these reactivity coefficients it is necessary to add considerable amounts of "moderators" (B_4C, BeO, etc.). It should also be noted that the more Np is existing in the fuel the more Pu-238 will be produced during transmutation (this isotope brings up problems for re-fabrication and reprocessing).

In the area of reactor safety trend analyses on the CAPRA core behaviour were made considering the initiation phase of a ULOF accident and using the newest version of the SAS 4A code. The results indicate a tendency for the CAPRA core design for an enhanced margin to sodium boiling onset when compared to the EFR core design. The results of post-failure transient analyses also show a favourable behaviour of the CAPRA core during core destruction. The potential for highly energetic power transients seems less pronounced than in the EFR design.

Within the CAPRA project SIMMER-II is used as an analysis tool for the core meltdown simulation and the recriticality energetics assessment until SIMMER-III becomes fully available. First analyses have been performed for large diluted cores and for Uranium-free cores with a Pu enrichment of approx. 40 %. The recriticality energetics in such cores does not deviate significantly from the energetics in traditional cores because the impact of the reduced Doppler-effect is compensated by other mitigating effects. However, in the case of strong separation processes of the Pu-fuel from the matrix/diluent material the energetics increases. Further analyses are underway to investigate the transition phase behaviour of such cores and to reduce the uncertainty levels of the calculation.

Besides the irradiation in the French SILOE and PHENIX reactors a common experiment (TRABANT) will be performed in the HFR Petten. The purpose of the experiment is to study the behaviour of the various new CAPRA fuel types up to medium burnup levels. In the first step of the experimental series 3 different fuel pins containing

- mixed oxide with a high content of PUO_2 (45 at%)
- mixed oxide with a high PuO_2 (40 at%) and Np (5 at%) content
- Uranium-free fuel types (PUO_2/MgO and PuO_2-CeO2)

will be tested.

2.5.2 Chemical separation processes

Already for some years (1991-93) studies have been made for procedures to separate the long-lived actinides from the fission products in the high-level wastes of the PUREX process. The aim of this work was to develop a flowsheet for a solvent extraction process and the main finding of the last years was that all extractants containing only sulphur are not suitable for the separation of transplutonides (III) from lanthanides (III).

In the next step (1993) we concentrated on mixed nitrogen/sulphur extractants (aliphatic shift bases) which had two nitrogen and two sulphur donor atoms in the molecule. But also with these compounds no tendency was found to distinguish between for instance Am (III) and Eu (III).

Therefore, in 1994 extractants which only contain nitrogen donor atoms are investigated (aromatic shift bases or Diamine with analogous chains but without double bounds). It should be mentioned that the synthesis of the starting extractant compounds are very difficult and some purification steps are necessary to refine the product. So, the first distribution experiments could be made with small amounts only, but the first results indicate that - depending on the acidity of the solution - the long-chained diamine extractants give better separation values than the aromatic shift bases.

2.6 Cladding and Wrapper Materials

In 1994 extensive investigations were made on pressurized tube samples irradiations in PFR (Experiment PFR M2) concerning the swelling and in-pile creep behaviour of this austenitic steel type DIN 1.4970 (= 15Cr 15 NiTi) up to a high dose of 125 dpaNRT. The best results were obtained for the cold-worked variation with a relatively high Si-content and a slight understabilization. Swelling and in-pile creep are closely related so that the latter can be determined from the first, at least below 600°C. The results confirm our opinion that this material is very much suited for fuel element application in Fast Reactors up to very high burnup levels.

In a specific fuel pin irradiation (KNK II-VENKER) it could be shown that the temperature gradient in the cladding will increase swelling by as much as 50 % compared to material irradiations; this is a consequence of he He-bubbles migration in the temperature gradient and leading to an accelerated void nucleation.

The investigation of a martensitic wrapper, made from DIN 1.4914 with head and foot from AISI 315 and irradiated in PHENIX to a maximum dose of 105 dpaNRT (Experiment SAMARCANDE) showed that the deterioration of the mechanical properties is rather small only and that both the element design and the martensitic material are appropriate for applications as a wrapper for FR fuel elements.

STATUS OF FAST REACTOR DEVELOPMENT IN INDIA

S.B. BHOJE
Reactor Group,
Indira Gandhi Centre for Atomic Research,
Tamil Nadu, India

Abstract

The economic liberalisation process has accelerated the industrial growth and this requires considerable energy input. Nuclear energy has to play an important role in supporting the increasing demand of energy. In this year FBTR was operated at 10.2 MWt power in a sustained manner. Several physics and engineering experiments were carried out. The mixed carbide fuel has achieved a peak burnup of 16,000 MWd/t without failure. First batch of irradiation experiments was completed and pins were delivered to RML for PIE. For PFBR, a thorough review of the conceptual design was carried out towards reducing the capital cost, construction time and for improving plant reliability. A 2 loop concept with 2 PSPs, 4 IHXs and 2 secondary loops having 4 integrated SG modules has been finally chosen with a expected savings of 15% in NSSS capital cost, 2-3 years in construction time and 10-12% in capacity factor. Number of materials to be used for major components was reduced to three to facilitate speedy development. Operating temperatures were finalised after optimisation studies. Discussion is in progress to finalise fuel handling system. Research and development activities are continuing at IGCAR in the areas of reactor physics, engineering development, core engineering, thermal hydraulics, structural mechanics, metallurgy, post irradiation examination, instrumentation and electronics, chemistry, fuel reprocessing, safety research and health physics etc.

1.0 BACKGROUND

The economic liberalisation process started in India since 1991 is continuing and has yielded good results during the last few years. During the fiscal year 1994-95 the GDP increased by 5.3% and the industrial production by 8.7%. The growth in the capital goods manufacture is very impressive at 20% and exports have increased by 17% while imports grew by 20%. There has been an impressive growth in the foreign exchange reserves which climbed to 20.9 b US $ in March 95 from a level of 1 b US $ in 1991. It is very well realised that the industrial growth is possible only when adequate energy resources are made available, particularly electricity. Due to financial constraints, the installed capacity is not growing as required. The Government policy on privatisation has resulted in the capital flow from abroad for installation of coal and gas fired power stations. Proposals are being discussed for laying gas pipelines from middle-east to India. Renewable energy resources are getting an impetus due to its environmental friendliness, simple technology, short gestation periods and external assistance.

The installed electricity generation capacity by the utilities in India by the end of Mar '95 is 81.2 GWe (Coal-52.2; Hydro-28.8; Gas-5.6; Nuclear-2.3; Oil-0.3). The total capacity addition during the current year is 3.5 GWe. The electricity generated is 351 TWh (Coal-234.0; Hydro-82.5; Gas & Wind-25.5; Nuclear-5.6; Oil-3.4). The cumulative addition of the capacity in the first 3 years of the current Five year Plan (1992-97) has been 11.6 GWe against 30 GWe planned. The target for the 9th Five year Plan (1997-2002) is to realise a capacity of 54 GWe in all the sectors. Major share in this is to be taken by the private sector.

With regard to Nuclear Power, India has commissioned its 10th power reactor, a 220 MWe PHWR unit II at Kakrapar in Gujarat State, in Jan '95. The unit I of the Narora Atomic Power Station, 220 MWe PHWR was connected to the grid in Jan '95 after rehabilitation following a fire accident in Mar '93. Several modifications have been carried out to prevent recurrence of such an accident. Two MOX fuel subassemblies have been loaded in Unit I of BWR Tarapur Station to demonstrate recycling of plutonium in thermal reactors. Fabrication of 4 more MOX subassemblies is in progress. The Tarapur units have completed 25 years of successful operation in Dec '94. China has supplied enriched uranium for the continued operation of Tarapur units. A long shutdown has been taken for Unit I of Rajasthan Atomic Power Station to replace the coolant channels. Construction of 2x220 MWe PHWRs each in Karnataka and Rajasthan states are progressing. Components for the 2x500 MWe Tarapur III & IV units are being received at site whereas the civil work is heldup due to shortage of funds. The total installed nuclear capacity is 2300 MWe and the cumulative experience gained is more than 110 reactor-years. Cumulative nuclear electricity generation, by all the units since beginning of the nuclear energy production in India till Dec '94, is 92.2 TWh.

Construction of India's third fuel reprocessing plant (KARP) of 100 t/a capacity at Kalpakkam is nearing completion and the plant is expected to become operational by the end of 1995. This will reprocess PHWR fuel discharged from nearby 2x220 MWe units of Madras Atomic Power Station.

The economic competitiveness of nuclear energy is decreasing due to long gestation periods and low capacity factors. Efforts are being made to improve these.

There is no change in the country's emphasis on the important role of nuclear power in meeting energy needs. The significant role to be played by FBRs in enhancing nuclear energy contribution is also well recognised. The parlimentary standing committee on energy has recently taken up its review of the role of FBRs in the energy options of the nation. R & D efforts are continued, mainly at the Indira Gandhi Centre for Atomic Research, for the operation and utilisation of FBTR and for the design and development of PFBR.

2.0 FAST BREEDER TEST REACTOR (FBTR)

FBTR is a 40 MWt/13.2 MWe, mixed carbide fuelled, sodium cooled, loop type reactor. It has been provided with two once through serpentine type Steam Generators (SG) in each of the two secondary loops. The reactor has 100% steam dump capacity, in order to continue reactor operation when TG is not available. The reactor achieved its first criticality in Oct '85. Since then, it is operated at various power levels in stages upto 10.2 MWt. The small carbide core with 25 fuel subassemblies, has been licenced to operate upto 10.5 MWt with 320 W/cm linear heat rating of fuel. TG will be commissioned shortly.

2.1 Reactor Operation

1994-95 was an eventful year for FBTR. During this period, high power physics & engineering tests were carried out, sustained operation at 10.2 MWt was achieved, irradiation of first set of test pins for Mark I & II fuel were completed, irradiation run with 24 fuel subassembly core was completed, 25th fuel subassembly was loaded and low power physics measurement were completed.

During the period Apr'94-Mar'95 reactor operated for 2,640 h with high power operation for 990 h (with SG) (Fig 1). FBTR has logged 13,250 h of cumulative operating time, out of which 1,900 h has been at high power. The fuel has seen a peak burnup of 16,000 MWd/t without any failure. An earlier conservative estimate of 25,000 MWd/t achievable burnup has been recently upgraded to 50,000 MWd/t after detailed analyses of FCMI.

During the year, reactor tripped 20 times (6 scrams and 14 LORs). In addition, there were 6 forced manual shutdowns, one planned annual shutdown and 7 planned shutdowns for high power physics & engineering tests. Required design modifications, replacement of components and strengthening of operation and maintainace procedures are progressively carried out to reduce the number of reactor trips. In addition to preventive maintainance, the major jobs carried out during the shutdown periods include fuel handling operations, insitu silicone bellows replacement on one CRDM, replacement of one bank of 220V UPS and 2 banks of 24V DC batteries for neutronic instrumentation and reactor protection system, installation of cover gas hydrogen meter on one secondary sodium loop for detection of leak in SG and ultrasonic examination of secondary sodium loop welds (refer Fig 1).

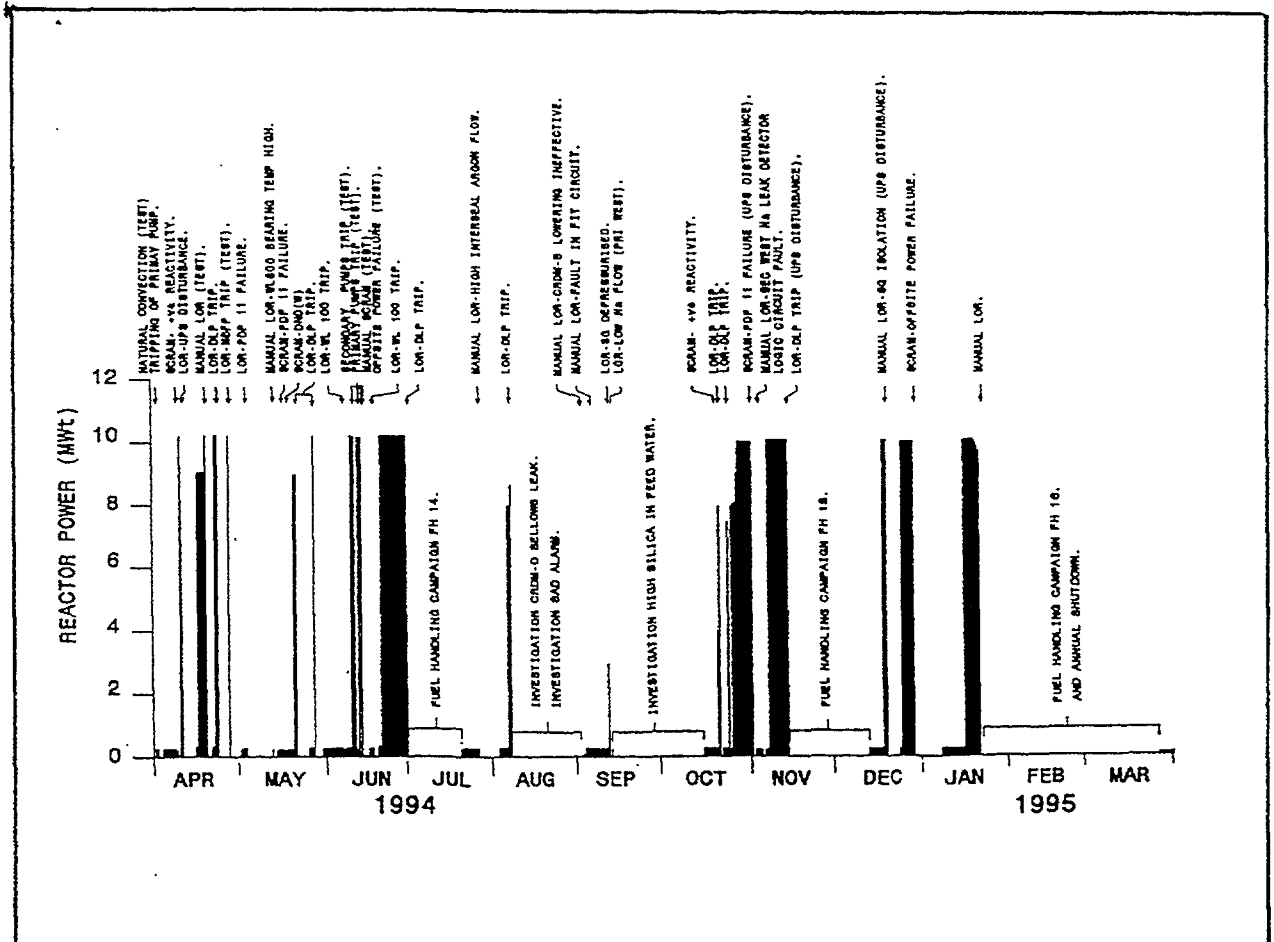


FIG. 1. Histogram of reactor operation.

2.2 Physics and Engineering tests

Following physics and engineering tests at various power levels upto 10.2 MWt were carried out successfully and the results were as predicted.

- Natural convection tests by tripping primary sodium pumps at 160 kWt
- Flux tilting experiments at 8.5 MWt
- Reactor kinetic measurments at low & high power upto 8 MWt
- Manual LOR test from 10.2 MWt for incident analysis
- Feed water pump auto take over and trip incident with safety action at 10.2 MWt
- Measurement of power coefficient of reactivity at low & high power upto 9 MWt
- Reactor shutdown by one primary sodium pump trip from 10.2 MWt
- Reactor shutdown by one secondary sodium pump trip from 10.2 MWt
- Manual scram test from 10.2 MWt
- Offsite power failure test from 10.2 MWt

2.3 10.2 MWt operation

During the year, reactor was operated at 10.2 MWt intermittently. 10 d sustained operation at 10.2 MWt was achieved for the first time in June '94. The performance of reactor systems, sodium systems, CRDMs and other safety related systems and auxiliary systems were generally satisfactory. Reactor could not be operated for longer periods at high power due to problems faced in maintaining the stringent water quality requirements of once through SGs. These were overcome and all the chemistry parameters were brought within the stipulated limits during the power campaign in Jan '95. A new 3 cu.m/h capacity demineralised water plant has been installed.

The heat transport parameters achieved during 10.2 MWt operation are as follows.

• Reactor inlet/outlet temperatures	:	330/420°C
• Primary sodium flow	:	349 cu.m/h
• Central fuel subassembly sodium outlet temperature	:	485°C
• Sodium temp. at SG inlet/outlet	:	418/290°C
• Secondary sodium loop flow	:	136 cu.m/h
• Feed water temperature	:	190°C
• Feed water flow	:	15.3 t/h
• Steam temperature	:	416°C
• Steam pressure	:	117.5 Kg/sq.cm

2.4 Irradiation of experimental fuel pins

Fuel pin irradiation programme to enhance the fuel performance in terms of burnup and LHR has been started. 3 Experimental fuel pins of Mark I (70% PuC-30% UC) and Mark II (55% PuC-45%UC) fuel compositions were irradiated and discharged from the reactor for post irradiation examination.

2.5 Performance of various systems

The primary and secondary sodium purity has been maintained below the plugging temperature of 105°C. The sodium pumps and their drives are operating very well and have logged more than 66,000 h.

Augmentation of radiation shielding at various hot spots in reactor building have been completed and significant reduction in dose rates have been achieved.

In addition to the existing fourth generation computer, a second computer UNIPOWER-30 developed indigenously has been commissioned. Software validation and testing of switchover logic have been completed and it is kept in auto mode. These computers supervise the core temperatures, check the health of safety logic systems and perform data logging.

2.6 TG commissioning

All the works connected with TG rolling have been completed except for achieving required vacuum in main condenser. Leak paths were identified and works are in progress to arrest them. Clearance has been obtained for TG synchronisation with grid and this activity is planned to be taken up shortly.

2.7 MARK II Core

Safety report has been prepared for the MARK II core (55% PuC-45% UC) which will produce 40 MWt power with 76 fuel SAs. Discussion on this with Regulatory Authority, is planned during 1995.

3.0 PROTOTYPE FAST BREEDER REACTOR - DESIGN

Preliminary design of 500 MWe, pool type, sodium cooled Prototype Fast Breeder Reactor (PFBR) was carried out in 1985. The concepts selected for the PFBR at that time were 4 Primary Sodium Pumps (PSP), 8 Intermediate Heat Exchangers (IHX), 4 Secondary Sodium Loops & Pumps (SSP) and 36 Steam Generator (SG) units. These concepts were based on the design trends of large size FBRs at that time. The construction experience of BN-600, SPX-1 and MONJU indicates that FBRs have high capital costs compared to PWRs. The construction experience of 220 & 500 MWe PHWRs in India has shown that the interest during construction (IDC) and escalation during construction (EDC) contribute significantly to the Unit Energy Cost (UEC) of electricity. Interest rates are high in India (12-18%) and construction periods are longer (8-10 years). Therefore, short construction periods are absolutely essential for acceptability of the nuclear power. The pace of construction of PHWRs is reduced recently due to paucity of funds and therefore, the construction of PFBR is being postponed by a few years. Considering all the above three factors, it was decided in 1993 that systematic efforts are to be made to reduce the capital cost of PFBR. Considerable expertise has been developed in analytical capabilities in the field of reactor physics & shielding, thermal hydraulics, structural mechanics and safety analysis during last 5 years. Experimental facilities are being set up for thermal hydraulics & structural mechanics studies, testing of components in sodium and development of special instrumentation for sodium systems.

In the improved design of PFBR, the number of components have been reduced drastically. However the overall systems & component concepts have been kept the same as in intial 4 loop concept.

3.1 Design Objectives

- The purpose of constructing PFBR is to demonstrate on an industrial scale techno-economic viability of an FBR Power Plant in India. The cost of electricity from PFBR should be comparable to that of PHWR and coal fired power stations. This requires considerable reduction in the capital cost of PFBR which in turn reduces the construction time and thereby IDC.

- The concepts selected for PFBR should be based on the operating experiences of FBRs. Any innovative design concept is to be incorporated only after thorough research and Prototype test. Atleast 5 reactors of 500 MWe capacity are likely to be built after PFBR. Therefore, the designs must be optimised and standardised so that the follow on plants will have minimum modifications.

- High breeding ratio is not an essential requirement of PFBR Fuel cycle. The growth of FBR initially will depend on availability of technology, availability of finance, public acceptance and then on Plutonium. PFBR will be a breeder reactor but without emphasis on high breeding. Economics of fuel cycle is an important factor at this stage of FBR development.

- Experience in design, construction and operation of all the FBRs, particularly all the incidents and accidents, shall be considered systematically in PFBR design.

- The reactor shall meet the PFBR safety criteria issued by AERB.

- Design simplification i.e. reduction in number of systems and components without compromise on safety and reliability, is to be carried out. Experience of thermal reactors indicates that complex plants take longer times to construct and are difficult to operate.

3.2 Design criteria

Parallel to the conceptual design the following design criteria are being defined.

- Safety criteria - already issued.

- Identification of applicable design and construction codes for all the systems and components, such as mechanical equipment of NSSS, electrical systems, instrumentation & control systems, buildings and structures, conventional equipment and plant siting is being done. For mechanical components of NSSS, ASME Sec III, Code Case N47 and RCC-MR are being considered.

- All the components have been categorised into four safety classes depending on the effect of failure on safety and cost.

- Seismic categorisation of the systems and components has been done, indicating the loading level classifications for SSE & OBE, appropriate stress limits and functional requirements during and after seismic events.

- Design Basis Events, their frequencies and acceptance limits during such events have been compiled.

- PFBR will be located at Kalpakkam site on the coast of Bay of Bengal. It will be a part of the Southern Region Grid which will have a capacity of 35 - 40 GWe by the year 2005. Therefore, shutdown of the rector will not have a large impact on the grid. The reactor will be operated as base load station, with provisions for long periods of operation between 25 to 100% power. The reactor will participate in the load changes in pre-determined manner but will not follow the load changes in the grid. It will not participate in the frequency control of the grid.

- The plant is being designed for 30 years life with 75% capacity factor.

3.3 Selection of Main Options

3.3.1 Core

After prolonged discussion on the choice of fuel, MOX has been decided for PFBR because of extensive experience, in India and abroad from thermal power reactors; excellent performance in all FBRs with very high burnup capability and proven reprocessing technology. The core is homogeneous and has two fuel enrichment zones. The fuel pin dia is 6.6 mm with 217 pins/SA. Core height is 1 m and diameter is 1.97 m. Spacer wire is used for locating the pins. The upper and lower blankets made of depleted U oxide are integrated in the fuel pin. The overall length of the fuel pin is 2.7 m and SA length is 4.5 m.

Enriched B4C is used as absorber material in control and safety rods (9 nos) and shut off rods (3 nos).

3.3.2 Number of components and loops for Na systems

The high capital cost of FBR is attributed to the complex technology coupled with increased number of primary & secondary sodium loops and associated components. Despite the disadvantage due to higher cost compared to established reactor systems (PWR, PHWR and BWR), the technical maturity and undisputable strategic advantage of FBR has led to rethinking on its design to enhance the economic competitiveness without compromising on overall safety and reliability aspects so as to gain acceptance as major energy producing option.

It is worth noting that the investment cost is the major contributor (~75%) of the UEC in comparison with 15% and 10% for O&M and fuel cycle cost respectively. This indicates that the major efforts should be directed towards reducing the capital cost which inturn calls for a lesser number of systems/components. Construction times will also reduce with lesser number of components. This has a significant advantage in terms of lower IDC and EDC. Another factor which plays an important role in the economy is capacity factor. From PWR data, it has been recognised that more number of components, apart from increasing capital cost, tend to reduce the plant availability because of more unplanned outages and more plant maintainance time. One more fact which has been inferred from the operating experience is that the (n-1) loop operation or reduced power operation has not contributed significantly towards plant availability. Hence increasing the number of loops for the sake of part loop operation will not yield overall cost benefit. To summarise, FBR is being designed with the minimum number of systems/ components from economy point of view.

For the primary sodium circuit, 2 PSPs and 4 IHXs have been selected for PFBR. With 2 PSP design, seizure of pump (if only one pump is in operation) would result in the clad hotspot temperature exceeding the allowable limit of 1173 K. Hence it is decided to shutdown the reactor in case of fault in any pump and replace the pump and then resume operation with 2 primary pumps. As a matter of philosophy, a minimum of 2 loops are essential so that in case of non-availability of one loop, the decay heat could be easily removed by other loop. PFBR has four independent safety grade decay heat removal loops consisting of DHX dipped in hot pool with heat rejected to air in AHX. It is logical to use secondary circuit for DHR with off-site power available. It is assessed that 2 loop concept will give higher capacity factor because of high reliability of no components except SG, short replacement time for pump when needed. Adoption of reduced number of loops reduces the amount of piping, layout space for components and time for their installation at site.

Considering the advantages of simplification in design, manufacture, construction schedule and operation, availability of proven turbine steam reheat cycle has been finalised. As regards to number of SG modules, operating experience on single wall SG has shown that this component plays an important role in plant load factor. Studies on operation with (n-1) SG, in case of a small leak indicates that minimum number of SG units/loop should be 3 in order to permit acceptable temperature dyssymmetry in the reactor cold pool. The temperature assymmetricity in the cold pool is caused because of the differences in the temperatures of primary sodium outlets from IHXs of affected and unaffected loop. The choice of 4 SG/ secondary loop has been arrived at based on preliminary optimisation analysis of capital cost and outage cost of a leak, with due consideration to construction schedule while permitting (n-1) SG module operations. 4 secondary loops with 2/3 SG units/ loop is not favoured because of high plant cost due to IDC.

The 2 loop concept with 2 PSP, 4 IHX and 2 secondary loops each loop having 1 secondary sodium pump and 4 integrated steam generator modules, has been decided for PFBR. The saving in overall cost of NSSS is about 15%. The construction time is expected to reduce by 2 to 3 years. Based on PWR experience, the increase in capacity factor could be 10-12%. Many PWRs up to power levels 600-700 MWe have been built with 2 loops. Fig 2 shows the compactness of the component layout for the case of 2 loop concept, comaparing with the 4 loop concept. Fig 3 shows the simplification of inner vessel and roof slab for the 2 loop concept.

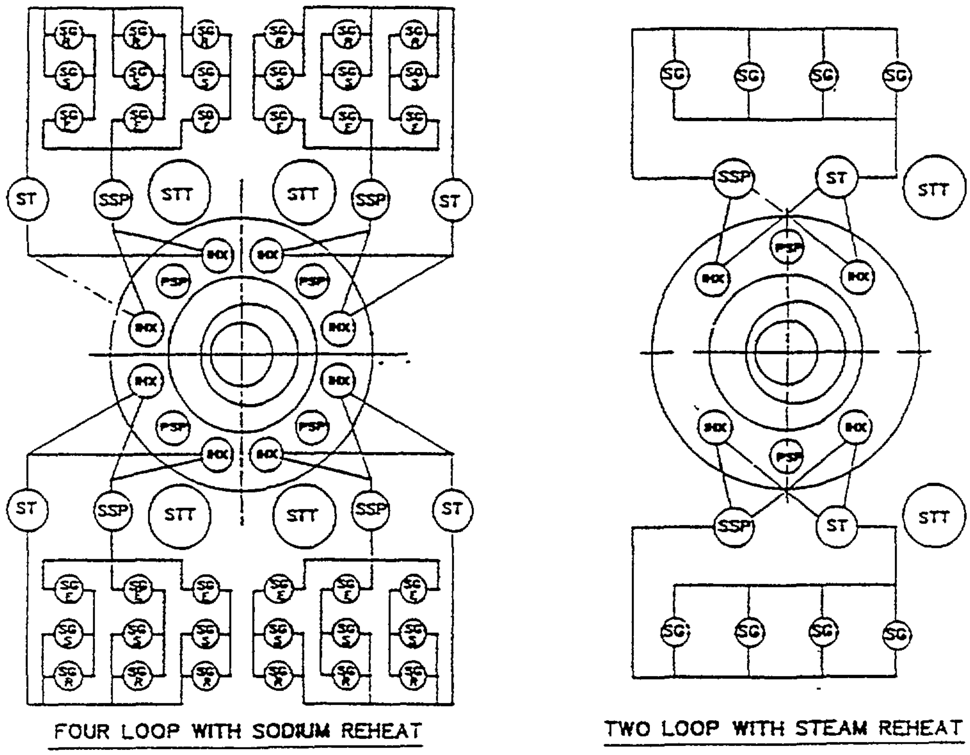


FIG. 2. PFBR component layout.

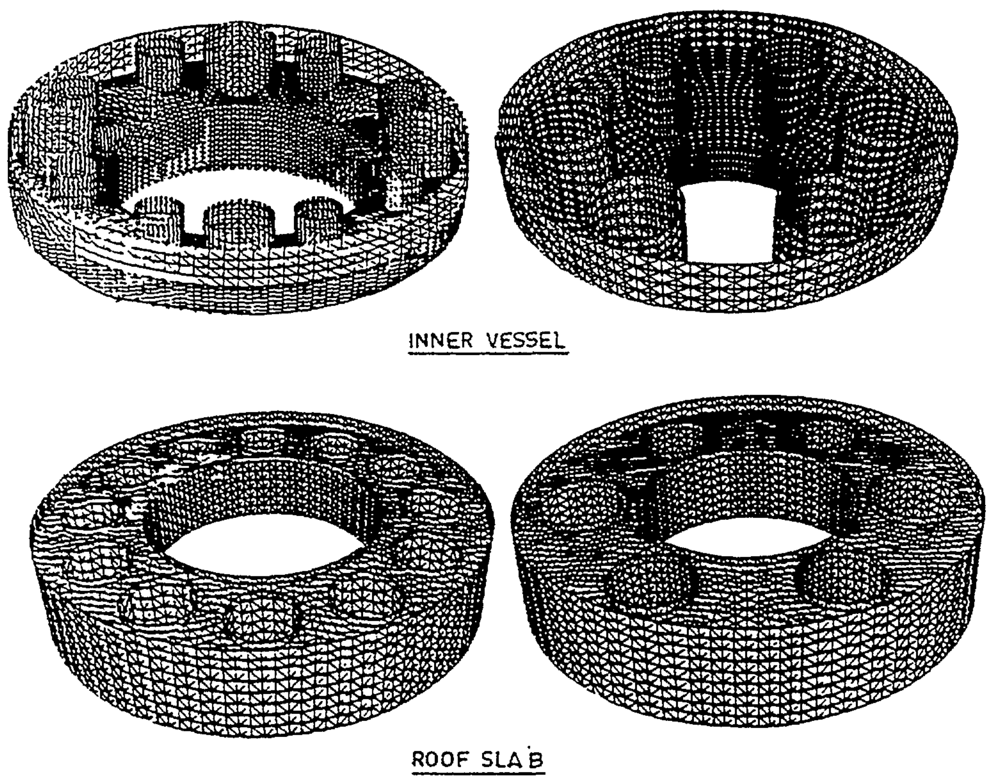


FIG. 3. Simplification of components.

3.3.3 Choice of main materials

20% CW D9 material is selected for cladding and hexcan because of its improved resistence against swelling due to neutron irradiation, high strength at operating temperature and good corrosion resistance against Na and fuel. SS 316 LN is selected for out of core components except for SG. Use of SS 304 LN for cold leg sodium components is being discussed. Modified 9Cr-1Mo steel has been selected for SG because of its adequate high mechanical strength, freedom from the risk of stress corrosion cracking (problem with stainless steels) and also decarburisation (problem with 2.25Cr-1Mo). Carbon steel ASTM A516 is selected for roof slab & rotatable plugs.

Discussion on materials for steam-water circuit is in progress. Titanium has been selected for condenser tubing because of its enhanced resistance against sea water corrosion and extensive world wide experience.

Physical and mechanical properties of all the materials required for the design, are being compiled.

3.3.4 Operating Temperatures

High reactor outlet temperature is preferred for achieving high thermodynamic efficiency. However, this is limited by the fuel burnup and component structural integrity considerations. In order to satisfy the allowable clad hotspot temperature of 973 K, the reactor outlet temperature is to be limited to 833 K for core ΔT of 150 K. As regards structural integrity of high temperature components, with the recent advancements in high temperature design codes and structural analysis methodology, it is possible to select as high as 825 K for the reactor outlet temperature. Detailed inelastic and viscoplastic analyses have been performed for the control plug, inner vessel and IHX using ORNL and Chaboche models. While elastic analysis route permits the reactor outlet temperature of 775 K in order to satisfy the rules of RCC-MR, the viscoplastic analysis indicates that the temperature can be 825 K. Modified 9 Cr-1Mo the structural material for SG can permit upto 770 K steam temperature. The turbines used in the conventional thermal power stations allow steam temperature upto 811 K. The reactor inlet temperature and hot & cold temperatures of the sodium in the secondary sodium circuit are arrived at from overall cost optimisation studies. Based on all the above considerations and discussions with the supplier of turbine for steam reheat cycle, the following plant temperatures have been selected (see Fig 4).

- Core inlet/outlet : 670/820 K
- Primary sodium inlet/outlet to IHX : 817/667 K
- Secondary sodium inlet/outlet to IHX : 628/798 K
- Feed water inlet/steam outlet : 508/766 K
- Steam conditions : 763 K at 17 MPa

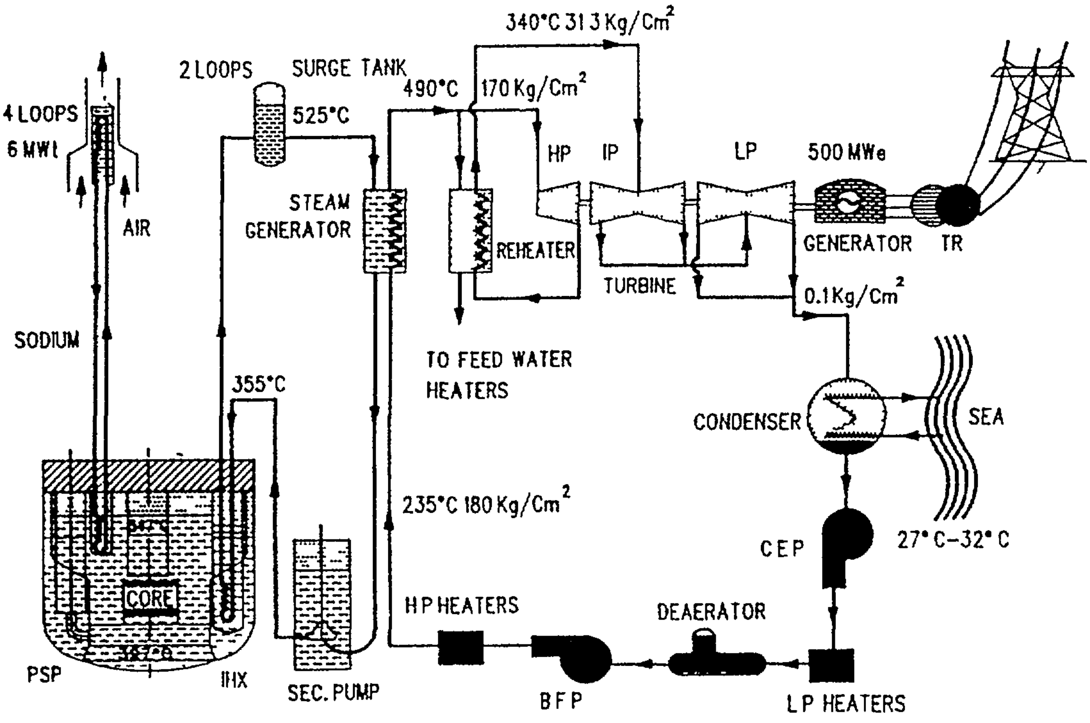


FIG. 4. PFBR flow sheet.

3.3.5 Core component handling

Analysis of operating experience on core components handling systems indicate the following:

- Fuel handling time in FBR is not governing the duration of shutdown of a plant Therefore, parallel operation of fuel handling machines is not needed (10-12 d/a shutdown for fuel handling, 40-50 d/a shutdown for maintenance)

- Though fuel handling equipment are more and complex in FBR compared to thermal reactors, their reliability is high. Incidences have taken place in good numbers but one gets back quickly

- Reduction of rotatable plug diameter is essential if advantage of reduced number of PSP & IHX is to be taken for reducing main vessel diameter. For this the following are essential

- reduce diameter of handling area- reduce blanket sub assembly rows, reduce the number of shielding subassemblies and locate invessel transfer position in the area of shielding subassemblies

- by combination of number of Rotatable Plugs (2/3) and types of IVTM (Straight Pull Machine(SPM)/Transfer Arm(TA) and their numbers).

The following four concepts are under discussion

- 2 rotatable plugs with 1 TA
- 2 rotatable plugs with 1 TA + 1 SPM

- 2 rotatable plugs with 2 SPM
- 3 rotatable plugs with 1 SPM

Irradiated fuel is stored in main vessel for one refuelling interval (180 EFPD) and exvessel storage is under water.

3.3.6 Primary Sodium Pump (PSP)

The concept of single stage, single suction, free sodium level, mechanical pumps has been retained for 2 loop option. The size of the sodium pump has been reduced drastically by reducing the margin on NPSH, decreasing the head to be developed and by increasing submergence of the impeller. The size of the 2 loop pump is about 1800 mm compared to 2100 mm in case of 4 loop earlier option. A critical review of the flow halving time resulted in requirement of about 8 s. It is possible to mount a flywheel directly on the PSP shaft. Non-return valves are eliminated as they are not required from core cooling considerations during transients. They can reduce the reliability of the pumps and increase the cost. One pump operation will not be carried out. In case of any major problem, the pump will be replaced with a spare pump.

3.3.7 Intermediate Heat Exchanger (IHX)

4 IHXs have been decided for the 2 loop concept. Conventional shell and tube concept with secondary sodium down comer at the centre and single pass concept has been retained. With appropriate primary hydraulic arrangement to minimise the radial temperature gradient, it is found that a bend in the tube is not necessary. The size of the IHX is reduced by reducing diameter of the tube from 21/23 mm to 14/16 mm. By increasing the pressure drop on the primary sodium side from 1 to 1.5 m and also by adoption of mechanical seal between the inner vessel and IHX, the overall size of the IHX is reduced. The ratio of PSP to IHX, 1:2 is being retained based on the large experience on such a concept and controllability of Na flows through the IHX.

4.0 RESEARCH AND DEVELOPMENT

4.1 Reactor Physics

A new coupled neutron-gamma multigroup library for 26 nuclides has been generated, based on Evaluated Nuclear Data File ENDF/B-IV for shielding calculations and validated. Activation cross sections for a number of materials used as activation foils for neutron flux and spectrum measurements have been prepared.

Development of a 3D diffusion theory burn-up code 3DB for studying the changes of fuel composition with burn-up is in progress. A two grid acceleration scheme has been developed and incorporated in a 2D hexagonal geometry code. Anlysis of leak noise data for SG obtained from IAEA under Co-ordinated Research Project is in progress.

Design Basis Accident analysis codes PREDIS and VENUS have been validated against European LOFA benchmark problems. Fuel subassembly worths at different radial positions in core during initial fuel loading were calculated. A study was made on the possibility of recriticality of molten fuel dropped in the core catcher from the core following an accident. Calculations of neutron irradiation dose for the reactor assembly out of core componets both in radial and axial locations were completed and indicated that the dose values are negligible (<<1dpa).

4.2 Engineering Development

The main activities under engineering development are setting up of Large Components Test Rig (LCTR), component hydraulics, sodium pump and instrumentation, amongst others.

4.2.1 LCTR

LCTR having a sodium inventory of 84 t has been commissioned. The necessary experimental conditions for the study of heat and mass transfer in the test vessel are being achieved. Fig 5 shows the schematic of test vessel set up for the study of heat transfer in the control plug.

4.2.2 Sodium Pumps

Sodium pump development test rig has been designed to study the dynamic behaviour of the rotating assembly of PSP and to verify the performance of critical components such as hydrostatic bearing, mechanical seals, coupling, spherical seat etc. by experiments.

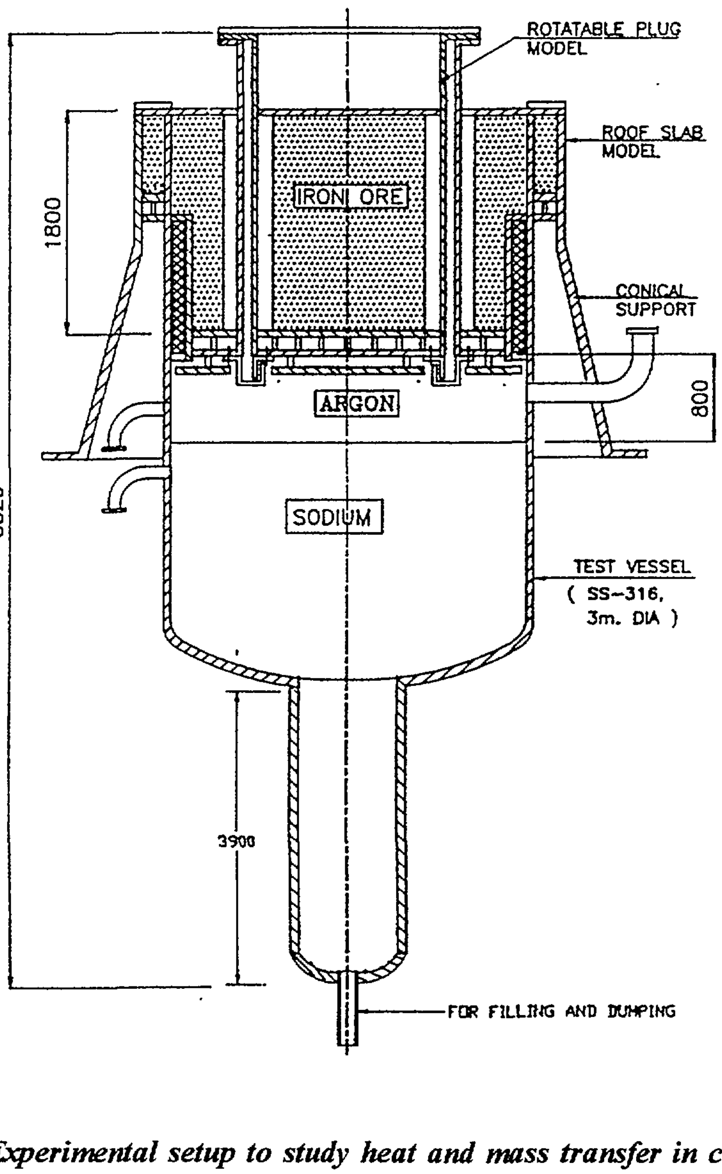


FIG. 5. Experimental setup to study heat and mass transfer in covergas.

4.2.3 Testing of Control & safety Rod Drive

Tests on EM have been completed. Further tests with the design modification are being done. Experimental setups are being fabricated for dash pot and seals. For the shut off rod drive, a full scale model of EM immersible in Na has been tested in air to demonstrate the load carrying capacity with the available space. Na Vapour deposition studies in vertical annuli of CRDM are being carried out.

4.2.4 Instrumentation

Steam leak signal data received from IAEA for developing noise analysis technique to detect and localise the leak source was analysed using the cross correlation and cross power spectral density methods. The transit time delays between the leak source and the different sensors were estimated based on the above methods.

4.2.5 Boron enrichment plant

Construction of a boron enrichment plant with a capacity of 5kg/a of 90% enriched B10 is in progress.

4.3 Core Engineering

A full scale dummy fuel assembly for hydraulic testing has been fabricated at Nuclear Fuel Complex, Hyderabad. This will be tested at IGCAR in the hydraulic test rig which is being installed. The core plan was finalised taking into account number of in-vessel storage locations required and additional shielding that may be required for invessel transfer post. The temperature limits for fuel cladding tube under transients were determined from the data available on transient tests and the limits have been stipulated as 1073 and 1173 K.

4.4 Thermal Hydraulics

4.4.1 Theoretical Studies

Maximum clad temperatures during seizure of one PSP and PSP trip, have been estimated (Fig 6) and found to be less than the corresponding allowable limits for D9 material (1073 K & 1173 for level B & level C loading respectively). Steady state temperatures were estimated for the integrated SG which need to be used for the tube and shell buckling investigations. In this, the effects of sodium by pass and plugging of about 5 % tubes in the central and perpheral zones have also been studied. The difference between average tube bundle temperature and maximum tube temperature is less than 27 K. The variation of temperature in the cold pool in the vicinity of IHX outlet is found to be less than 20 K and the corresponding variation in the main vessel bottom region is less than 5 K. In order to reduce the cold thermal shock on the hot pool components under reactor scram, studies were done on the thermal transients following a reactor scram with PSP trip as a sympathetic safety action. In order to limit the clad hotspot temperature less than allowable value of 1073 K during a total power failure incident, the minimum primary flow halving time needed for PSP has been determined as 8 s.

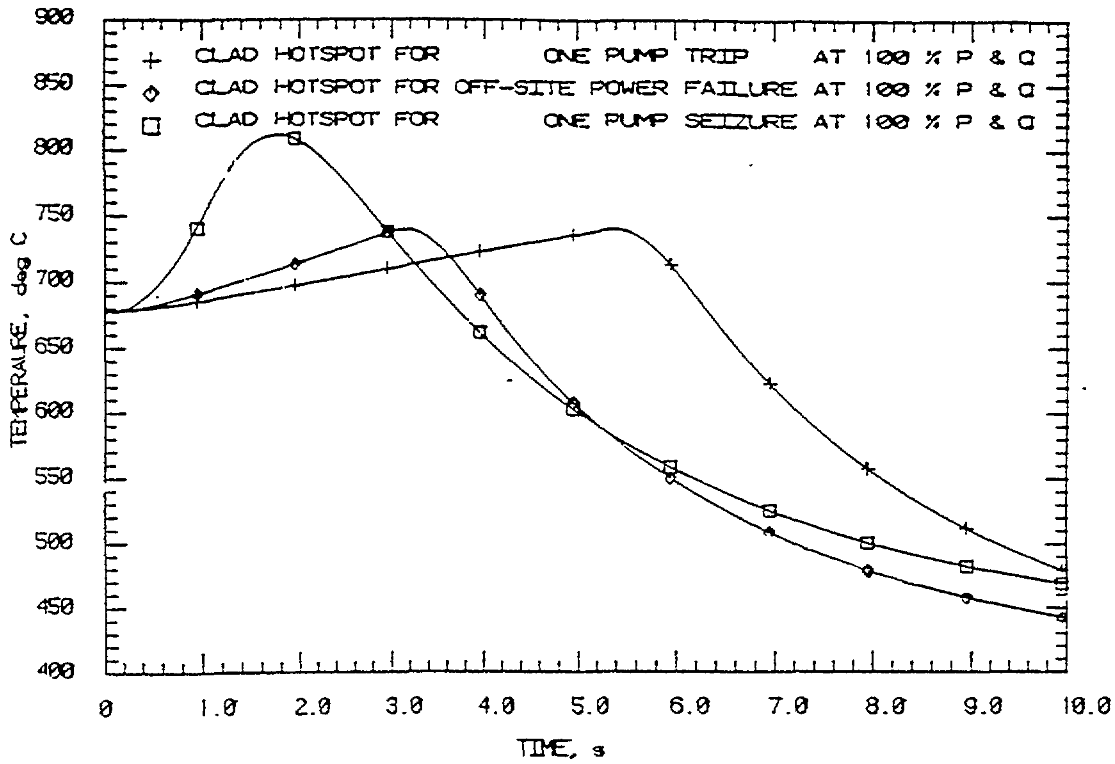


FIG. 6. Two primary pump PFBR concept - primary pump incidents with scram on power to flow ratio at 1.25.

4.4.2 Experimental Studies

An experimental study was carried out in water on a semi-transparent 1/3 scale, 90 degree sector model of main vessel to confirm even temperature distribution along its circumference and also to ensure the freedom from FIV risk. Velocity measurements were also made in the above model using a miniature sized 3 mm dia propeller anemometer. The grid plate experiments in 1/3 scale model in air have been started in collaboration with Fluid Control Research Institute, Palghat (Fig 7). The objectives of the experiments are to study the flow and pressure distribution with and without the baffle plate at different operating conditions, to select an optimum baffle plate configuration if required and flow visualization. Thermal stratification studies were continued on the 1/24 scale model of hot pool simulating conditions after a reactor scram.

4.5 Structural Mechanics

4.5.1 Theoretical Studies

Detailed elasto-plastic buckling analysis of inner vessel for 2 loop concept was completed including shape optimisation studies. The optimised geometry of the inner vessel can permit upto 1.5 m of differential sodium pressure for the through wall temperature gradient of 90 K. This result is useful for deciding the IHX dimension. For the study, the effect of creep on the buckling strength of inner vessel has also been investigated. Based on creep buckling analysis of one of the shells of IHX and also the analysis of inner vessel, it has been concluded that creep buckling failure mode is not of concern for PFBR components. It has also been inferred from the analysis that Creep Cross Over Curve given in RCC-MR is applicable for creep buckling failure mode also. The effect of imperfection on buckling of main vessel is studied for

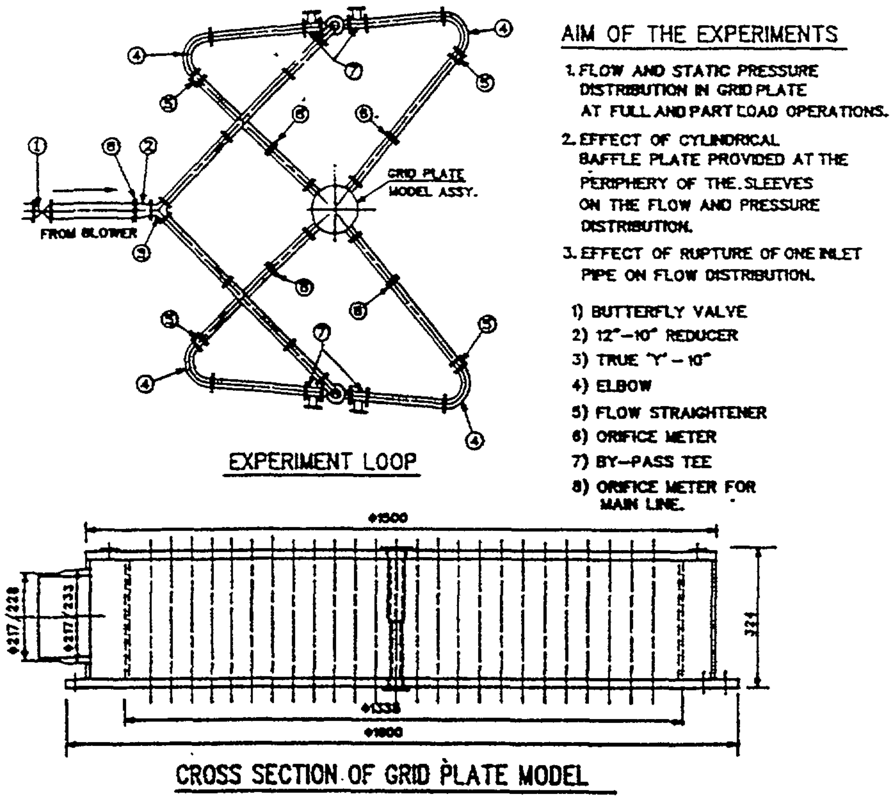


FIG. 7. PFBR grid plate experiments in 1/3 model with air.

the seismic load of 0.078 g OBE (S1) and 0.156 g SSE (S2). The results have indicated that, the straight portion with 25 mm thickness will meet the RCC-MR buckling rules for the maximum imperfection of 35 mm, whereas the dished end can permit the maximum imperfection of 12.0 mm at the knuckle portion.

Towards improving IHX tubesheet design, two important aspects namely the effect of tube hole pattern and optimisation of the profile of the tubesheet shell junction are studied. An improved finite element model with the use of an higher order finite element to account for the shear effects in the IHX tubesheet analysis has been identified. The profile of the top tubesheet shell junction has been improved by attaching the shell closer to the neutral axis of the tubesheet and reducing the excess mass at the rim. This has shown better thermomechanical behaviour under thermal transients. Analysis has indicated that, with improved profile at the junction, the IHX can satisfy the 'elastic' rules of RCC-MR for the temperature of 825 K. The detailed seimic analysis of IHX by response spectrum method has been done including the FSI effect, for both S1 & S2 seismic excitations. Parametric studies have been done for understanding the effect of modelling of tube bundle and pipe mass. The functional requirement is met as there is no mechanical interaction between IHX and inner vessel during S1 excitations. The maximum stress intensity meets the RCC-MR stress limits.

A straight tube concept is being considered for SG due to many attractive features, like simplicity, ease of fabrication, less fabrication cost, no fear of stress concentrations and elastic follow up issues. However the only drawback with this is the risk of thermal buckling. Detailed theoretical buckling analysis has been performed including parameteric studies on span length and radial clearance at intermediate support locations. Analysis indicates that, even without bends in the tubes and bellow in the outer shroud shell it is possible to avoid buckling risk upto the temperature gradient of of 50 K (an absolute temperature rise in the most critical tube with respect to the tube bundle/shell). This value is higher than the maximum value of 38 K predicted by the detailed thermal hydraulics study. Thus the study will be useful in selecting a straight tube concept for SG. However detailed analysis is in progress for verifying the structural integrity under transient operating conditions.

4.5.2 Experimental Studies

FIV measurements on a 19 pin bundle model of fuel subassembly were carried out in an hydraulic rig. The vibrations measured are well below the maximum permissible limit stipulated to prevent high cycle fatigue failure. Four buckling experiments, on 1/40th scale models of main vessel under shear load and external pressure, were performed (Fig 8). Comparison with theoretical prediction is satisfactory. A sophisticated test loop for assessing the structrual integrity of control plug mock up is being setup at IIT, Madras (Fig 9). All the fabrication and erection works have been completed and the loop will be ready for thermal simulations studies. Dynamic experiments on empty and water filled aluminum tanks under seismic base excitation to study the shell mode and beam mode contributions of the tank under seismic base excitation are done at Structural Engineering Research Centre, Madras. The experimental data on the natural frequencies of empty vessel have shown good comparison with the CASTEM 2000 code. Collaborative projects on Leak Before Break analysis for SG with IIT, Bombay and seismic analysis of reactor assembly at IISc Bangalore were started.

For conducting in-house static and buckling experiments, a 100 t test rig is being installed at IGCAR.

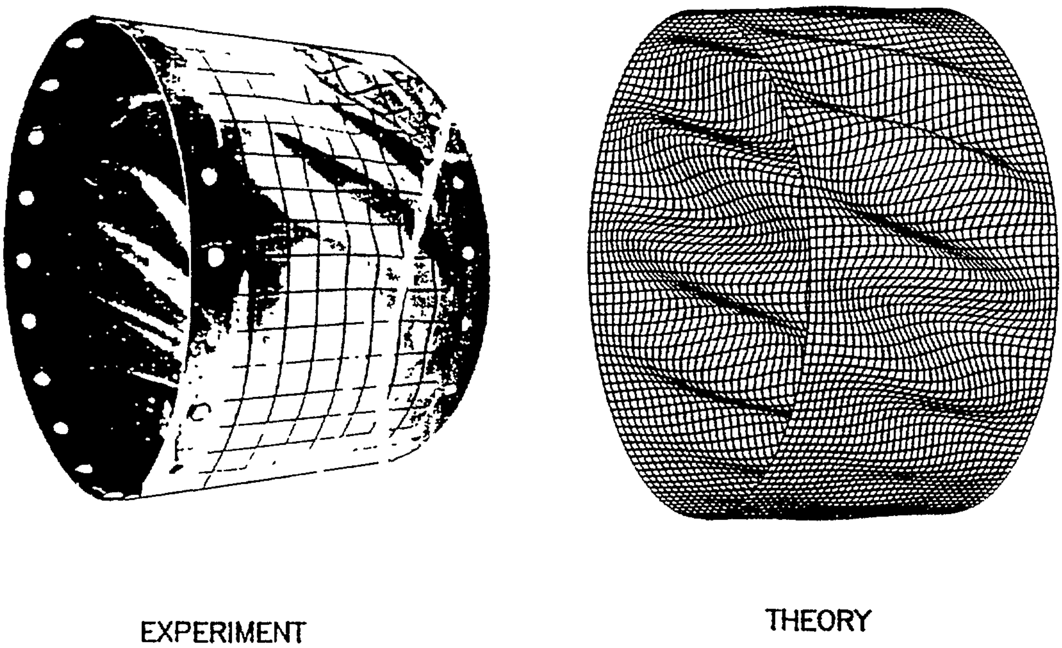


FIG. 8. Buckling modes SS model under shear load.

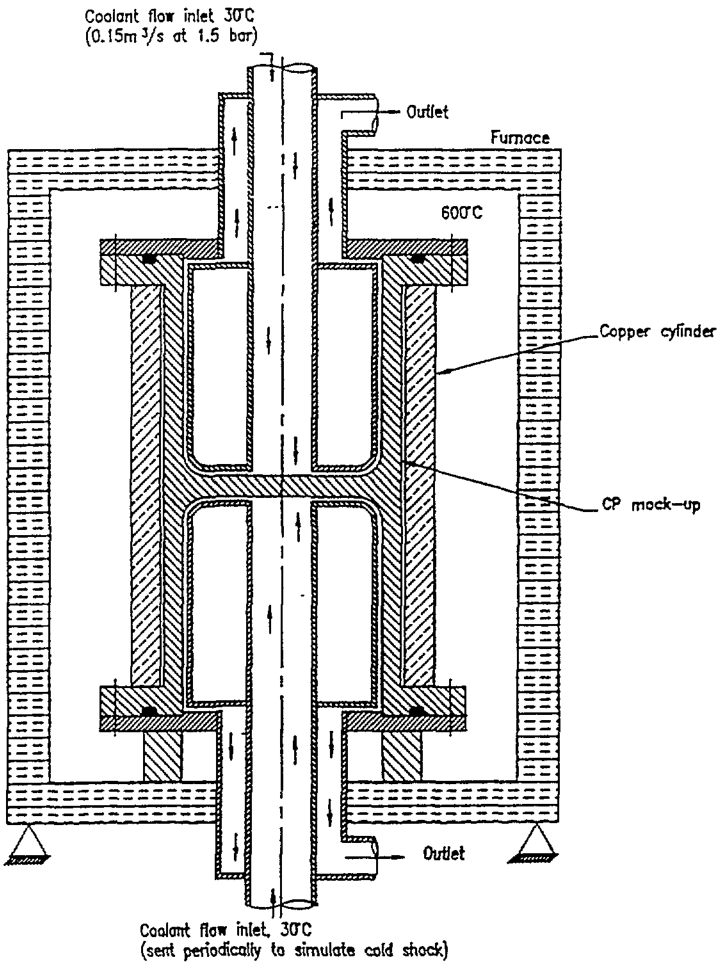


FIG. 9. Control plug mock-up test setup.

4.6 Metallurgy

4.6.1 Creep Properties

Creep tests were performed on all-weld and weldment specimens for SS 316LN with SS 316 electrodes at 923 K over a stress range of 115 to 245 MPa. Steady state creep rates and rupture strengths of the weldment were generally found to be higher compared to that of the weld metal. However, in the stress range of 115 to 145 MPa, the rupture strength of the weld metal was higher than the weldment. Failure of the weldment specimens always took place in the weld metal. In contrast, rupture ductility of the weldment was marginally less as compared to the weld metal. In comparison to base metal, ductility of the all-weld and weldment was very poor. Inferior creep properties of the weld metal are attributed to the transformation of delta-ferrite to carbides and sigma phase during the creep tests. Extensive cavitation was observed in association with the sigma particles.

Creep rupture behaviour of 2.25Cr-1Mo and 9Cr-1Mo ferritic steel base metals and welded joints were studied at 823K over a stress range of 100 to 250 MPa. No significant difference was noticed in base metal creep rupture strengths for both the steels, whereas rupture strength of 2.25Cr-1Mo weldment was inferior to that of 9Cr-1Mo weldment. Weldments generally exhibited poor creep properties compared to base metals. At a given applied stress, the difference in the rupture life of the base metal and weldment was more pronounced in 2.25Cr-1Mo steel than in 9Cr-1Mo steel; this difference increased with decrease in the stress level. Poor creep properties of weldments of 2.25Cr-1Mo arise due to the absence of Mo_2C carbides in the intercritically heated heat affected zone (ICHAZ) and hence creep failure invariabally occurs in ICHAZ. On the other hand, strength in 9Cr-1Mo is due to solid solution and sub-structure development which is relatively less sensitive to degradation by welding thermal cycles. The reduction in the creep rupture strength of weldments compared to base metal in 9Cr-1Mo is, therefore, less pronounced. However, failure again occurs in ICHAZ.

4.6.2 High Temperature Low Cycle Fatigue Properties

Low cycle fatigue (LCF) tests were conducted on thick section 9Cr-1Mo steel tube plate forging at frequencies in the range of 0.01 to 1 Hz with strain amplitude of ±0.5%. The stress response was characterised by a small amount of initial hardening followed by progressive cycling softening. Fatigue life was found to be nearly constant at all the frequencies; the variation was within ±10%.

Studies on creep-fatigue interaction behaviour of SS 316LN stainless steel base metal, carried out at 873K with hold times in the range 1 to 90 min, have clearly established the deleterious effect of tensile hold on fatigue life as compared to that of either compression hold or tension+compression hold. Development of creep cracks and cavities and propagation of cracks assisted by oxidation were identified as the main factors responsible for decreasing the fatigue life. A comparison of the crack density as a function of hold time and type of hold revealed that intergranular crack density was very less in compression and tension+compression hold tests, while crack propagation was mainly intergranular in tension hold tests. Crack linkage was observed to be more with increase in the length of the hold time in tension.

4.6.3 Microstructural stability at elevated temperature

Influence of cold work on the microstructural stability and the recrystallisation kinetics of Alloy D9 was studied in the temperature range 873 to 1123 K. Specimens with prior cold work (15 to 22.5%) were isothermally heat treated. The kinetics of recrystallisation was determined from the variation of fractional softening as a function of time at temperature. Activation energy for recovery and recrystallisation was found to be maximum for 20% cold work thus implying that the microstructure is the most stable for this level of cold work. Therefore, the suggested optimum cold work to resist recovery and recrystallisation at high temperatures is 20%.

Investigations on the high temperature microstructural stability of several compositions of SS 316 weld metals have been carried out. The results from these studies have clearly indicated that delta-ferrite transforms by a two-stage process. In the first stage the ferrite transforms mainly to carbides and austenite while in the second stage the transformation products are sigma and austenite. It was also observed that the most significant elements controlling sigma phase formation are C, Cr and Mo. While C showed a strong inhibiting effect, Mo was about four times as effective in promoting sigma than Cr.

4.6.4 Weldability

Hot cracking susceptibility of 316LN and Alloy D9 was studied using the Moving Torch Varestraint test method. The extent of cracking in type 316LN alloy was more than that could be ascribed to a primary austenitic solidification mode, ferrite potential and impurity element concentration. This indicated an interaction between N and P in synergistically promoting cracking. Cracking in Alloy D9, however, conformed to that of fully austenitic materials and exhibited features typical of Ti-added austenitic alloys. The HAZ in the two materials was prone to backfilling. The backfilling of hot cracks was a function of thermal conditions, composition and capillary dimension.

4.6.5 Process maps

The constitutive flow behaviour of SS 304L has been studied in the temperature range 293 K to 1523 K for the strain rates varying from 0.001 to 100 1/s with a view to establishing processing-microstructure relationships during hot, cold and warm working. The efficiency of power dissipation through microstructural changes is plotted as a function of temperature and strain rate, to obtain a processing map. The different domains exhibited by the map are correlated with specific microstructural processes occurring during hot working. There exists a peak efficiency of 33% occurring at 1423K at 0.1 1/s. which are the optimum conditions for hot working of this material. In this domain the ductility reaches a maximum value and the grain size variations permit considerable scope for microstructural control. It is preferable to cold work the material at strain rates higher than 0.1 1/s and to avoid warm working. The process maps developed for the commercial SS 304 have shown more or less similar flow instability regime. Studies were also repeated for the cast SS 304 and SS 316L. The large scale validation experiments involving press forging, rolling, hammer forging and extrusion have confirmed the predictions of the processing maps. These experiments have demonstrated that the processing maps can be used in industries to control the microstructure of the component and to reduce the rejection rates during deformation processing.

4.6.6 Fracture mechanics

Evaluation of dynamic fracture characteristics of 9Cr-1Mo ferritic steel was carried out for normalised & tempered (N&T), aged (at 1013 K for 1-2 h) and welded conditions (with PWHT). The drop-weight nil-ductility transition temperatures (T_{NDT}) were 248 K, 248 K and 268 K respectively for N&T, aged and welded conditions. Despite the difficulties in crack-profile measurement and the uncertainties in fracture loads due to microstructural variation in HAZ of the drop-weight specimens, the particular procedure developed enabled determination of conservative estimates of K_{Id} at/below T_{NDT} from instrumented drop-weight tests which are found to be 70 and 58 MPa_m for the N&T and weld materials respectively. These compare well with those determined from precracked Charpy tests. The ageing treatments employed here produced only marginal changes in toughness and transition temperature characteristics of N&T 9Cr-1Mo steels. For the welds, conservative K_{Id} values were obtained by applying novel data reduction procedures to load-time data from instrumented impact test of unprecracked Charpy V-notch specimens. Even the lowest value of these K_{Id} estimates in the tempearture range 193-303 K is found to be higher than the ASME K_{IR}.

4.7 Post Irradiation Examination(PIE)

PIE facility is intended to handle, among others, plutonium rich carbide fuel which necessitates inert atmosphere to be maintained in the hot cells. The leak tightness of the hot cells was checked and neccessary gasket and seals were incorporated to achieve the leak tightness below 2% of cell volume per day. After achieving the leak tightness to the desired level, the cells were filled with nitrogen in phased manner and the oxygen level was maintained well below 2%.

The hot cell facility was commissioned, with nitrogen atmosphere, in Dec '94 and two irradiated experimental fuel pins were received in a special steel subassembly from the spent fuel storage area. The receipt operations were carried out remotely using the leak tight laCalhene vertical transfer system. No increase in the radiation field was measured in the operating area as the pins were irradiated at a linear power of 120 W/cm for about 12d only and the cell shielding ability is very high. Subsequent measurement using the in cell gamma monitor indicated contact dose of the order of 1000 R/h from one pin. One experimental fuel pin has been removed from the capsule and is presently being examined using the NDT test facilities viz. X radiography, Eddy current test, and leak test facilities provided in the cells.

A X-Radiography system has been installed at an isolation area. This system consists of a miniature trolley for the transportation which is designed in such a way to have lesser distance from the fuel pin and film. The X-ray tube is hung from a trolley having facility to adjust the travel and centre height to vary the distance of X-ray tube from the fuel pin, to take radiograph of the full fuel pin or a portion of interest. Radiography of first experimental fuel pin was carried out in hot cell 3.

4.8 Linear Heat Rating Experiments in FBTR

The aim of the experiments is to study the evolution of structure of the fuel in reactor, to determine the maximum permissible linear heat rating and determine the swelling characteristics of the fuel. Experiment capsules for carrying out the fuel irradiation experiments in FBTR were fabricated. The capsules contain specially fabricated fuel pins which contains fuel pellets of the same composition as is presently used in FBTR. In addition, some of the fuel pins contain pellets of fuel which has MARK II core fuel composition. Irradiation of the first batch of experimental fuel pins has been completed. During irradiation, the temperature of the sodium exiting from the special subassemblies holding the irradiation capsules was monitored after lowering the core cover plate of the reactor.

4.9 ISI

4.9.1 Remote Field Eddy Current Technique (RFECT)

For the ISI of small diameter and thick 1Cr-1Mo (ferromagnet material), RFECT is an efficient technique. For this, a state-of-the art instrument has been developed. This RFECT instrument essentially consists of a low frequency oscillator, a lock-in-amplifier and allied circuitry. Special V-notch filters have been used to suppress the interferences from line frequencies and signal optimisation is achieved with the use of high-pass and low-pass filters. The instrument has demonstrated, repeatable and unambiguous detection of wall loss down to 10% wall thickness in 9Cr-1Mo steel tubes.

4.9.2 UE of Weld Joints of Secondary Sodium System

Welded joints of secondary sodium system of FBTR were subjected to Ultrasonic Examination (UE) as per ASME Sec XI, the purpose for which is to generate base line data for the future in-service inspection of these joints. Reference pipe standards having the same material and weld configuration for different pipe outside diameters were made. Rectangular notches having a depth of 10% wall thickness, 25.4 mm length and 3 mm width were made by milling in the heat affected zone both in the longitudinal and circumferential directions. Scanning was done from both the sides and in both axial and circumferential directions in order to cover heat affected zone and the entire volume of the weld. With this UE were conducted for welded joints of the secondary sodium system of FBTR, no unacceptable defect indications were observed.

4.9.3 Residual Stress Measurement in a Hydrostatic Pump Bearing

In order to protect the PFBR hydrostatic pump bearing surfaces from wear, hard facing is done by depositing suitable hard alloys (stellite or Colmonoy) either by welding or spraying processes. It is important that pump bearings must be free from residual stresses before doing the hard facing operation, as presence of residual stresses may lead to premature failure. In order to get base line data, residual stress measurement were carried out on a 300 mm outer diameter stainless steel cylindrical job before the hard facing operation. Hole drilling strain gauge method was used and the measurements were carried out as per ASTM standard procedure. The results indicated that along circumferential direction around 19 MPa (tensile) residual stress was present. The residual stress along longitudinal direction was found to be around 1 to 1.5 MPa (compressive). It is found that this small amount of residual stress present in the component before the hard facing treatment may not be detrimental. Residual stress measurements will be carried out again after the hard facing operation.

4.9.4 Metallurgical Examination

A technique called "Convergent Beam Electron Diffraction" has been standardised for the fingerprinting of various secondary carbides that evolve in nuclear grade steels. This technique would help to distinguish two similar carbides like Mo6 C and Cr23 C6 using their property called symmetry of the lattice.

The formation of dichromium nitride phase in 316LN (780ppm N) nuclear grade stainless steel has been investigated by TEM methods. It is found that on ageing the steel, in the temperature range 1023-1123 K for time durations up to 100h, precipitation of Cr2 N phase occurs. The precipitation of Cr 2 N phase has been reported for the first time, in such low nitrogen bearing nuclear grade austenitic stainless steels.

For an assessment of the microstructural stability of 9Cr-1Mo steel, precise data on the thermodynamic activities of Cr and C as a function of service temperature are needed. The adaptation of high temperature flourine concentration cells with CaF as the electrolyte for studies on this alloy has paved the way for the direct measurement of Cr activity as a function of temperature. Thus the chromium activity could be reproducibly determined in 9Cr- 1Mo steel for the first time and is found to lie between 0.054 and 0.076 in the temperature range of 900 to 1100 K.

Samples of SS 316 stainless steel, exposed to flowing sodium at different temperatures, were examined for chemical and mechanical property changes. The threshold oxygen contents in sodium for the formation of NaCrO when in contact with 316SS, 304SS, D9 alloy, 2.1/4Cr- 1Mo and 9Cr-1Mo steels were calculated. The influence of selective leaching of chromium on the carbon activity of sodium in the loop was assessed by a thermodynamic model.

Resistance to hydrogen induced SCC (HISCC) of 9Cr-1Mo steel, as evaluated by CERT method, indicated that the the HISCC resistance increases with increase in tempering time. Untempered steel is most susceptible to HISCC. Threshold stress intensity factors for HISCC and crack growth rates were established for this steel in various heat treated conditions in NaCl solution and aggressive acidic solution containing a recombination poison, As2 O3.

4.10 Instrumentation and Electronics

Simulation studies were carried out to detect the cause of distortion in the temperature distribution during gradual shutdown (LOR) of FBTR during 10.2 MWt.operation. The analysis led to the identification of sluggish thermocouple in the reactor core (response time greater than 6 s). The fluctuation in the core temperature signal was found to increase with reactor power. On-line software for the detection of plugging in fuel subassembly is being modified to take into account the inherent fluctuation in the process signal. 'Mobile Data Acquisition System' involving 'Hand Held Computer' is installed at FBTR. The data collected in this computer by the shift operator as well as the data scanned by conventional data loggers are periodically stored in off-line PC for the later retrieval.

4.11 Chemistry

4.1.1 Thermochemistry:

The free energies of formation of some chromium carbides and ternary carbides of uranium with Cr, Mo and W were measured by the methane hydrogen equilibration technique developed earlier.

4.11.2 Pyrochemical Reprocessing:

A train of glove boxes housing facilities for carrying out studies on pyrochemical reprocessing schemes has been set up. An electrorefining cell, with associated facilities for preparation of salt mixtures, consolidation of the deposits, distillation of cadmium etc., has been set up, and commissioning of this facility is nearing completion. Electrorefining experiments have been taken up in this facility, and the recovery of uranium has been demonstrated. Quantitative studies are now in progress to determine the yield etc. The current-voltage relationship in the electrorefiner has been studied as a function of the concentration of uranium trichloride initially present in the molten salt, and the optimum concentration of uranium in the melt has been arrived at. Studies have also been carried out on the recovery of uranium from salt waste by treatment with Na-Cd alloys. Basic studies have been carried out to measure the thermodynamic properties of $CeSn_3$ and liquid tin-cerium alloys by employing the molten salt EMF technique.

4.11.3 Non-destructive Assay of Plutonium in process streams:

An On-line alpha monitor has been developed for assaying Pu in trace levels in process streams. This monitor has also been qualified for uranium streams.The neutron collar for the measurement of Pu in larger concentrations has been tested in dynamic mode, and is ready for deployment. Methods have been developed for the non-destructive assay of Pu in glove box waste, and a system has been made operational for measurements on waste generated at IGCAR. An optical fibre adaption to a commercial fluorimeter has been set up to demonstrate the technique of Remote Fiber Flourimetry (RFF). This allows for determination of concentration of analytes, such as uranium, at remote locations. It has been shown that using organic ligands and micelles, fluorescence in the lanthanides can be enhanced by over four orders of magnitude. These ideas will now be applied to the poorly fluorescing actinides such as Pu and Am.

4.11.4 Reprocessing Chemistry:

A detailed study has been undertaken on the possible use of alternate trialkyl phosphates for reprocessing of fast reactor fuels by the Purex process. The extraction of U(VI), Pu(IV) and Th(IV) by a number of trialkyl phosphates was studied in detail, and these studies indicated that Triamylphosphate could be a good candidate solvent for fast reactor fuel reprocessing.

4.11.5 Sodium chemistry

A cover gas hydrogen meter has been designed and fabricated for this purpose to detect any steam leak into low temperature sodium in SG either during reactor start-up or low power operation. The meter was tested and calibrated in an experimental sodium loop by injecting hydrogen into sodium simulating the conditions of reactor sodium circuits. The meter has been installed in FBTR cover gas system and is being commissioned.

Electrochemical hydrogen sensors that make use of a mixture of CaH_2 and $CaCl_2$ as the hydride-ion conducting solid electrolyte were developed, tested and installed in FBTR. In order to minimise the drift during continous operation at 50 ppb level, the design of the meters was modified by reducing the thickness of iron membranes and increasing the area of the inner electrode.

Electrochemical carbon meter based on Li_2CO_3-Na_2CO_3 electrolyte with suitable design for incorporation into the reactor sodium circuits was fabricated and tested. In order to establish the effectiveness of in-sodium hydrogen sensors in detecting steam leaks into sodium at low temperatures (200°C) a series of steam injection experiments were carried out. While the injections at 200, 235 and 300°C gave measurable responses the responses to injections at 160°C were scattered, but negligible.

4.11.6 Solid State Chemistry

Surface oxidation behaviour of uranium carbide (UC) in air was studied. It is seen that formation of uranium oxide/hydroxide and free carbon can happen as a consequence of the reaction between UC and oxygen/moisture. It is noted that the surface of the oxidised UC consists of a top “contamination layer” of adsorbed oxygen/moisture, followed by a layer containing uranium oxide, uranium hydroxide and free carbon and then grain boundary oxide and finally bulk UC. The bahaviour of sintered and arc-melted samples is found to be similar. Thermal conductivity of mixed oxide fuel (45%U - 55%Pu) was measured using laser flash technique covering a temperature range from 700 to 1550 K.

4.11.7 Analytical chemistry and spectroscopy

An optical fibre adaptation to a commercial fluorimeter has been set up to demonstrate the technique of Remote Fiber Flourimetry. In order to understand the role played by fission product tellurium in the ful-clad interaction in FBR, the binary systems of tellurium with the alloying constituents of stainless steel namely Fe,Cr,Ni,Mo and Mn, were investigated by using high temperature mass spectrometry. The tellurium potential prevailing in the fuel-clad gap is insufficient for clad attack when the O/M is stoichiometric or below. However, clad corrosion is possible with hyperstoichiometric fuel. The first telluride that is likely to be formed is that of Mn followed by that of Cr.

4.12 Fast Reactor Fuel Reprocessing Plant

Preparation of cell equipment layout drawings is completed. Piping design is in progress. Process vessels fabrication is being tendered out. The design of auxilliary process system like makeup area, delay tank, high active storage system etc. are nearing completion. A mockup facility to test the viabilities of head and system is getting ready for commissioning. An air pulsed ejector mixer settler was developed and the hydrodynamics of the mixer was studied for flow capacities as low as 5 ml/min to 600 l/h. A 30 l/h, 6 stage unit has been demonstrated and a compact 20 stage unit for flow sheet studies is under fabrication. A new radially pulsed extraction column was proposed and the hydrodynamic and mass transfer behaviour of this unit was studied with HNOs - 30% TBP/n-dodecane system. An electrolytic thermosyphon dissolver for the dissolution of mixed carbide fuel is being manufactured employing titanium as the material of construction. The unit consists of platinum plated titanium as anode and titanium as cathode, both anode and cathode are separated by a ceramic diaphragm. Microwave direct denitration of Pu product solution to get PuO2 is being developed and a stainless steel microwave cavity separated from the micro wave generator, by means of a stainless steel duct for transmitting the micro wave energy has been fabricated.

Based on the R&D experience gained on various systems, a lead mini plant with a processing capacity of 1 kg/day for processing the FBTR fuel is in the advanced stage of erection.

4.13 Safety Research and Health Physics

Towards the preparation of radioactive source for the activity transport and deposition studies, method of preparation of electrodeposited source for Zn^{65} was standardised. A scale model of an experimental setup for wetbed submerged gravel bed scrubber has been made for assessment of its performance and exploring the feasibility of use of similar system in fast breeder reactor containment venting applications.

In connection with studies related to characterization of aerosols released during simulated reactor accidents, experiments are in progress on the physical and chemical properties of copper oxide aerosols generated by fast condenser discharge technique and it is planned to extend the same to nuclear fuels.

Studies were done on the recriticality potential of relocated molten core of PFBR towards assessing the use of poisons embedded in the core catcher as a means of enhancing the mass of molten fuel that can be supported without risk of criticality. The studies show that the traditional poisons such as boron or gadolinium are ineffective owing to the harder spectrum that prevails under the relocation condition.

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A REVIEW OF FAST REACTOR ACTIVITIES IN ITALY

R. TAVONI
ENEA, Energy Department,
Bologna, Italy

Abstract

The paper presents the progress of the work performed at ENEA for the optimization of the PRISM MODB oxide core in the framework of a collaboration agreement with General Electric. Studies on seismic isolation are outlined. The Italian contribution, mainly by industries, on SPX-1 restart is described.

1. ELECTRICITY PRODUCTION AND DEMAND

In 1994 the final consumption of electrical energy increased by 3% mainly as a consequence of a greater increase (+4%) in the industrial sector.

TABLE I - Electricity production and demand in Italy during 1993 and 1994/1/

	1993		1994	
	TWh	%	TWh	%
Thermoelectric Energy	174.3	78.2	180.8	77.9
- Oil derivatives	113.9	51.1	116.8	50.3
- Natural gas	39.6	17.8	40.4	17.4
- Solid fuels/2/	20.8	9.3	23.6	10.2
Primary Electricity	48.5	21.8	51.3	22.1
- Hydro	44.5	20.0	47.6	20.5
- Geo	4.0	1.8	3.7	1.6
- Nuclear	--	--	--	--
Total Gross Production	222.8	100.0	232.1	100.0
Net Electricity Import	39.4		37.6	
Total Availability	262.2		269.7	
Energy used by auxiliary production services and by pumping	- 15.6		- 15.9	
Network demand	246.6		253.8	

/1/ Provisional data
/2/ Coal, lignite and others

2. NUCLEAR RESEARCH AND DEVELOPMENT IN ITALY

Progress on LWR have mainly concerned the following activities:

- Westinghouse AP-600 reactor:

. Under a technical cooperation agreement among ENEL, ENEA, ANSALDO and WESTINGHOUSE, an integral test of AP-600 passive safety systems has been performed at the SIET facility in Piacenza.
The original test programme, including three AP-600 major design basis transients, small break loss of coolant accident, steam generator tube rupture and main steam line break, was concluded in October 1994. Two new tests were also performed to verify the facility repeatability and the effect of a more severe cooldown.
The Relap 5/mod 3 input deck developed by Ansaldo was sufficient to predict the overall system response, to reproduce reasonably the timing of the transients and the interactions between the safety systems.

. At the Vapore facility located at ENEA Casaccia Research Centre, testing of the Automatic Depressurization System (ADS) of AP-600 reactor coolant system is in progress.

- General Electric SBWR reactor

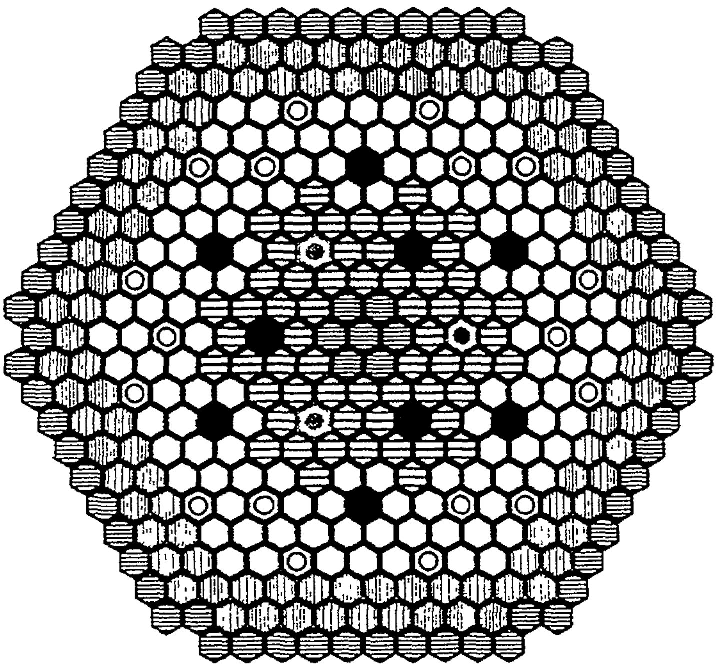
. The Passive Containment Cooling System (PCC) prototype supplied by ENEL, has been tested at the Siet facility in Piacenza. The confirmation that the design meets the thermal-hydraulic performance requirements for use in the GE-SBWR and is adequate to assure the structural integrity of the unit for the expected SBWR lifetime severe conditions, was reached.

An IAEA Technical Committee Meeting on "Progress in design, research and development and testing of safety systems for advanced water cooled reactors" will be held in Piacenza on 16-19 May 1995. During that meeting, organized in cooperatrion with ENEL, ENEA and SIET, a full presentation of tests results obtained worlwide is expected.

3. ITALIAN COLLABORATION WITH GENERAL ELECTRIC ON ALMR

- The optimization of safety parameters for a fast burner oxide core was presented at the International Topical Meeting held in Obninsk on October 3-7, 1994. Further studies have been continued with the aim of reaching the best burning capability of the PRISM MODB oxide core, satisfying the safety criteria.
A large number of core design parameters have direct influence on the above mentioned safety targets, but often in contradictory ways; therefore the optimization of the core from a global point of view is a very complex matter whose solution requires a sensitivity analysis based on a wide range of parameters including number and location of fuel assemblies, active core height, pin diameter, number and location of GEM's, Pu enrichments and their distribution, possible utilization of burnable poisons and their location, radial profile of the Na void effect, etc..

The core, which is now being studied to determine the transient characteristics, is shown in Fig. 1.
The new concept proposed by ENEA is based on the hypothesis of adopting two enrichment zones in the core: a lower Pu content for the internal zone, a higher Pu content for the external one. This proposal derives from the consideration that, during a fast transient, the largest increase of the Na temperature occurs in the peak power assemblies; therefore it should be advisable to have the power peak assembly where the Na void coefficient is minimum. Usually, the minimum values of Na void occur in the outer zone of the core, where the neutron leakage is higher; therefore, a power increase through higher enrichment in the external zone, can help to move the location of a possible Na boiling towards the region of more favourable Na feedback.



Low Enrichment Fuel	60
High Enrichment Fuel	120
Control	9
Ultimate Shutdown	3
Gas Expansion Module	18
Reflector	108
Shield	73
Total	391

Fig. 1 - ALMR Mod B Ox Burner Core Layout

Other peculiar characteristics:

- a central "hole" in the core corresponding to 7 positions, has been created to enhance the neutron leakage and to have a better Na void coefficient;
- no radial and axial blanket have been introduced to have less Pu production;
- the active core is enlarged (180 subassemblies) to compensate the reactivity loss with the 7 central shield elements and to have low rated fuel elements;
- a high number of GEM(18) is considered to achieve a better safety margin.

- A theoretical approach for the determination of the different quantities of tag gases to be introduced in the pin plenum, with the purpuse of identification of a subassembly with failed pins, has been considered. The selected tag gases

were Kr-78, Xe-124, Xe-126, Xe-128 with long irradiation half life compared to the assembly life, except for the Xe-124 (5.5 years half life with $\emptyset = 10^{15}$ $n/cm^2.s$).
This fact assures the conservation of the tag gas during the assembly life, and the short half life of the children makes it possible to reach an equilibrium concentration in a few weeks' or months' time. There are two limit conditions in which the minimum gas amounts have to be calculated: at the end of the assembly life (EOL) and at the last restart before EOL.
The most critical condition is at the last restart before EOL, when the tag gas is depleted, i. e. at the beginning of the last cycle.
A method to determine the different quantities of the gases to be inserted in different subassemblies in order to be possibly individuated by the detectors, has been proposed . Taking into account all different kinds of uncertainties, the optimization of the gas quantities to be employed can be reached.

4. SEISMIC ISOLATION

4.1. Guidelines development

A proposal for design guidelines for isolated nuclear facilities, has been prepared by ENEA with the cooperation of ALGA S.p.A., ANPA, ANSALDO-Ricerche S.r.l., ISMES S.p.A. and the University of Bologna in the framework of a EC Study Contract. It updates and extends a previous document which had been jointly published by ENEA and GE Nuclear Energy in 1990, and was limited to High Damping Rubber Bearing (HDRB).
The proposal applies not only to isolated reactors, but also to other nuclear isolated facilities (for instance, spent fuel storage pools).
The document has been published in a tentative form to allow for broad review from experts. It has been prepared taking into account the most recent information on seismic analysis of nuclear reactors in general and the state-of-the-art of engineering design of isolated structures.
The document will be periodically updated by ENEA to include comments and to reflect the advances in the seismic isolation technology development. When a satisfactory degree of consensus among experts is reached, the document will be submitted to Licensing Authorities for approval.
The qualification procedure specified for the isolation devices may lead to a definition of a standard and to the use of standardized products for seismic isolation design.
The document concerns all the horizontal isolation systems that may be suitable for application to nuclear structures, namely those formed by steel-laminated high damping elastomer bearings, lead plug bearings, low damping bearings with separate elastic-plastic or viscous devices, and sliding bearings; remarks are also provided on simultaneous horizontal and vertical isolation systems (3D systems).

4.2. Experimental activity on seismic isolation

Experiments in progress concern the use of high damping rubber bearing such as isolators, isolated structure mock-ups, and actual isolated buildings (in-site tests).

The above activities have been jointly continued by ENEA, ENEL,ALGA (which is the manufacturer of HDRBs in Italy), ISMES, the University of Bologna and ANSALDO Ricerche. With respect to previous studies, however, the analysis of the effects of different parameters affecting the HDRB behaviour, and their combinations, has been considerably extended: in particular, various types of attachment systems between bearings and steel end-plates (recess, bolts, central dowel, combined use of central dowel and bolts, and direct bonding), rubber compounds and shape factors have been considered. The most relevant recent activities which have already been completed or are in progress concern: (a) uniaxial, equibiaxial and shear tests on high damping rubber specimens; (b) static and dynamic tests of single isolators; (c) shake table tests of a half-scale isolated mock-up of electric equipment.
Uniaxial (tension and compression) and equibiaxial tests of rubber specimens have been performed at ENEL for both a rather hard rubber compound used in the U.S. Advanced Liquid Metal Reactor (ALMR) Project (shear modulus G = 1.4 MPa) and new medium hardness (G = 0.8 MPa) and soft (G = 0.4 MPa) compounds. Tests concerning the ALMR rubber have been carried out in the framework of a co-operation with GE Nuclear Energy.

4.3. Numerical models of HDRBs

Non-linear finite-element models of high damping rubber bearings have been developed and implemented in the ABAQUS code in the framework of Italian cooperative studies for seismic isolation development. The Hyperelastic models used have been based on the results of tests on rubber specimens. The isolators models are validated through comparison of numerical results with complete bearing test data.
The availability of reliable isolators models will allow the number of quite costly experimental tests on complete bearings to be considerably limited, not only as regards the analysis of the effects of some important parameters (e.g. temperature, ageing, vertical load on horizontal stiffness, etc.), but also for complicated experiments such as for instance, failure tests and analysis of the effects of defects.

5. SPX-1

- ANSALDO Divisione Nucleare has participated in both commissioning and service activities, with its own personnel on site, at Lyon (common team with FRAMATOME-Novatome) and at the headquarters in Italy.

  In such context, ANSALDO has developed or contributed to the following activities:

  - study of special reactor operating configurations, i.e. at low power with one loop down (out of four);
  - definition of guidelines for periodic testing of safety related equipment;
  - revision of the Safety Report to take into account the modifications of the fuel handling drum, the outcomes of the mixed sodium fire analysis and the results of commissioning tests;
  - participation in the "Crisis Team" with experts of reactor operation and structural mechanics;

- technical assistance for re-starting of the plant, mainly focused on intermediate (sodium-sodium) exchangers;
- design and implementation of those modifications on secondary loops, secondary tunnels and steam generator buildings related to protection against sodium fires; such activities have involved conceptual design, detailed design, definition of purchasing specifications, follow-up of supplier, monitoring of installation and commissioning of the modified parts;
- revision of emergency operating guidelines to take into account those implications deriving from the above modifications;
- thermodynamic analysis of the propagation of pulverized sodium fires through compartments;
- thermomechanical calculation of secondary piping and storage tanks in those environmental conditions deriving from sodium fires;
- leak before break analysis of secondary loops;
- design of modifications (process simplifications) on the fresh-fuel handling line;
- definition of the strategy of in-service inspection for some relevant components of the reactor (secondary loops, fuel cladding leak-detection and localization, in-vessel fuel handling machine);
- conceptual design and realization of the drive mechanism and control system of the steam-generator ultrasonic inspection device, a unique tool developed jointly by ANSALDO, FRAMATOME and CEA).

- EDF/NERSA commissioned Fabbricazioni Nucleari to supply 75 stainless steel reflector elements on October 27, 1994.
  Fabrication is under way and has to be completed by november 1995.
  A further supply of 175 reflector elements can be asked in the near future.

A REVIEW OF FAST REACTOR PROGRAM IN JAPAN

Power Reactor and Nuclear Fuel
Development Corporation,
Tokyo, Japan

Abstract

Development of the LMFR has progressed steadily in Japan with an aim of fast reactor commercialization by 2030. The total operation time of the fast reactor JOYO is about 52000 hours. The start-up of the prototype reactor Monju is in progress. The design study for a demonstration fast breeder reactor (DFBR) is progressing, and construction is planned to start at the beginning of the next century. Further R&D are being conducted on LMFR fuel technology and waste management. This paper describes the progress of R&D of the LMFR.

1. General Review

1) In accordance with the Long-term Program for Development and Utilization of Nuclear Energy defined by the Japan Atomic Energy Commission (JAEC), Power Reactor and Nuclear Fuel Development Corporation (PNC) is playing the key role in the development of a plutonium utilization system by fast breeder reactor (FBR), which is superior to the uranium utilization system by light water reactor, aiming to achieve future stable long-term energy supply and energy security of Japan.

2) The experimental reactor Joyo, located in the O-arai Engineering Center(OEC) of PNC, has provided abundant experimental data and excellent operational records attaining 51,078 hours operation in total by the end of March 1995, since the first criticality in 1977.

3) On the prototype reactor Monju, the initial criticality was achieved in April 1994 and the start-up tests are in progress. The reactor power has been increased gradually from February 1995. Generation of electric power and connection to the grid will be started in July 1995.

4) As for the demonstration fast breeder reactor (DFBR) of Japan, the Japan Atomic Power Company (JAPC) conducted conceptual design studies for the past several years, and confirmed the feasibility of top entry loop type reactor concept. Based on results of the design studies, the Federation of Electric Power Companies (FEPC) decided in January 1994 to start construction of the DFBR plant at the beginning of the 2000's. FEPC also decided the basic specifications of the DFBR plant.

The related research and development (R&D) works are underway at several organizations under the discussion and coordination of the Japanese FBR R&D Steering Committee, which was established by JAPC, PNC, Japan Atomic Energy Research Institute (JAERI) and Central Research Institute of Electric Power Industry (CRIEPI).

Progress of the design study and the related R&D are reported to the Subcommittee on FBR Development Program of JAEC.

5) Recent major emphases on the PNC's R&D are placed on the integrated feedback of all existing R&D results and experiences to the development of DFBR.

Furthermore, the overall functional and performance tests of Monju, is another important key role to attain further excellency of FBR technology, with full efficient usage of the test results.

6) R&D on following tasks are also in progress for development of DFBR, for excellent technology to attain FBR commercialization, and for technological breakthrough.

① improvement of reactor core safety
② improvement of plant safety
③ development of measures for probabilistic safety assessment
④ development of high performance core and fuel
⑤ study for various fast reactor core
⑥ development of structure and material for high temperature system
⑦ improvement of sodium technology
⑧ synthesis assessment of Monju data
⑨ synthesis assessment and improvement of plant technology using Monju
⑩ design study of plant system

7) In addition to the MOX fuel fabrication at the Plutonium Fuel Fabrication Facility for Joyo, Fugen (ATR), and BWRs in Japan, a new Plutonium Fuel Production Facility (PFPF) was constructed at Tokai Works of PNC. PFPF started production of initial core fuel of Monju in October 1989 and completed in January 1994.

8) On the FBR fuel recycling, adding to the experiences at the Tokai Reprocessing Plant, R&Ds are underway at three Engineering Demonstration Facilities (EDF-I, II, III) and

Chemical Processing Facility (CPF), integrating the results to the design of Recycling Equipment Test Facility (RETF) and future FBR Fuel Recycling Pilot Plant. The construction of RETF was initiated in January 1995.

9) Following the national program on waste management, PNC is also actively contributing to the area of vitrification of high level liquid waste, geological disposal of it, and low level transuranium bearing waste treatment, and promotion of construction of a storage engineering center in Hokkaido.

10) Aiming to the age of future FBR commercialization, further extensive and effective collaboration with foreign institutions will also have to play an important role.

2. Experimental Fast Reactor JOYO

2.1 General Status

This report covers the activities of JOYO from April 1994 through March 1995.

The 29th duty cycle operation and various experiments on the reactor characteristics and the plant performance were carried out successfully during this period. Figure 2.1 shows the core configuration for the 29th duty cycle. Major items of the experiments are as follows:

1) The burnup reactivity distribution
2) The coolant flow rate distribution in the core

The operating history of JOYO is illustrated in Fig. 2.2. The total operation time as of March 31, 1995 is 51,078 hours since the initial criticality in 1977, and also the accumulated thermal power is 4,097,357 MWh.

Many numbers of irradiation tests, such as for the nitride fuel and the carbide fuel, are now in progress. The creep test of fuel cladding material under irradiation with MARICO(Material Testing Rig with Temperature Control) was performed at the 29th duty cycle.

2.2 Upgrading Project of JOYO(MK-III project)

The MK-III project, which aims the upgrading of irradiation performance of JOYO, is now proceeding on schedule.

The safety review for the MK-III project by STA has been in progress.

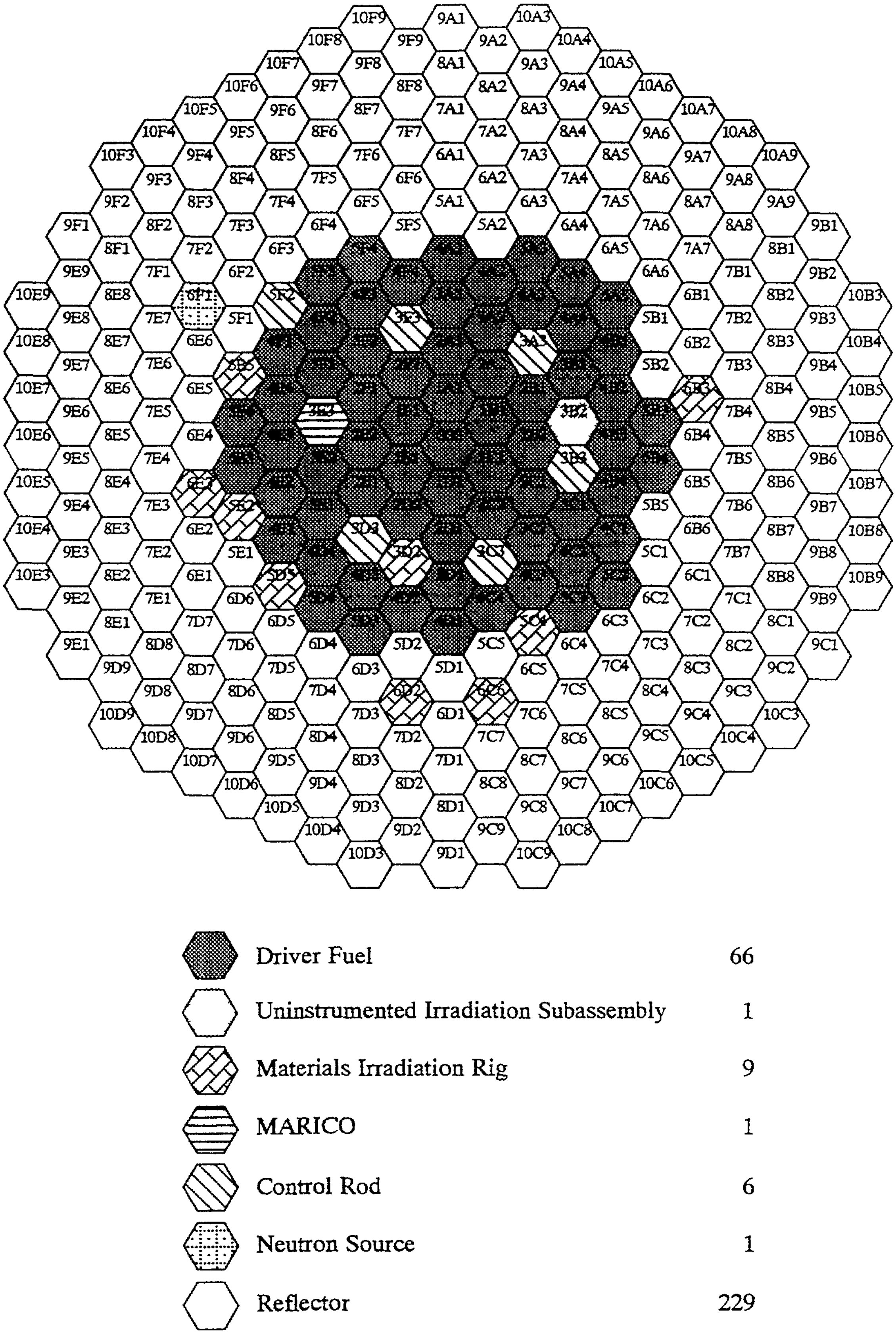
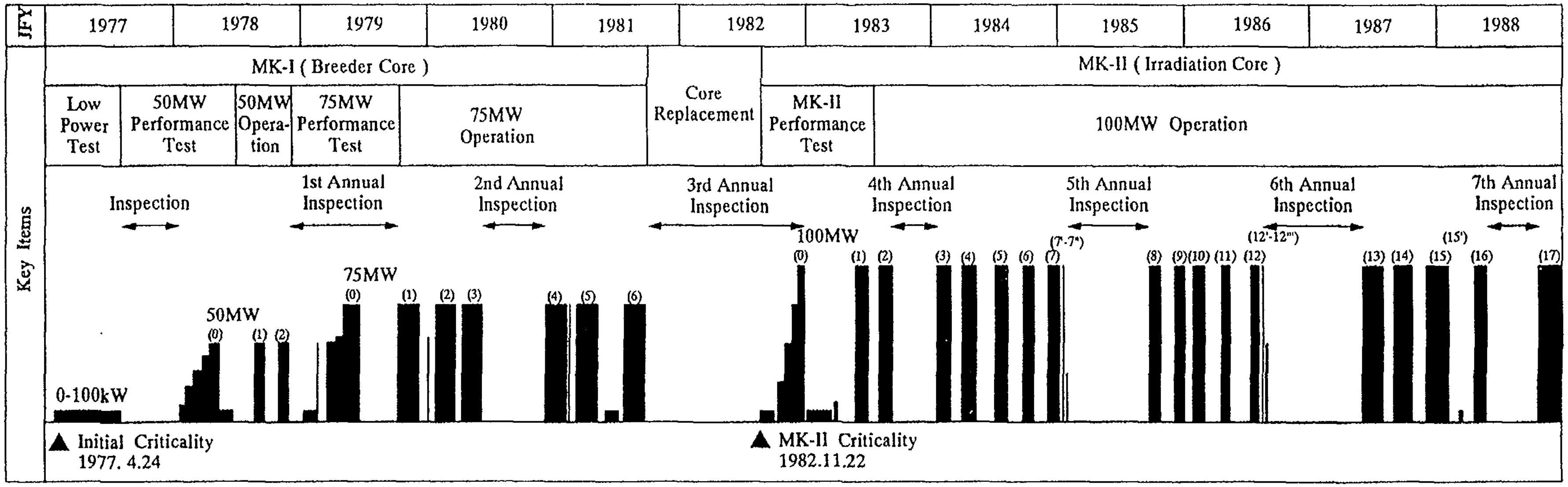
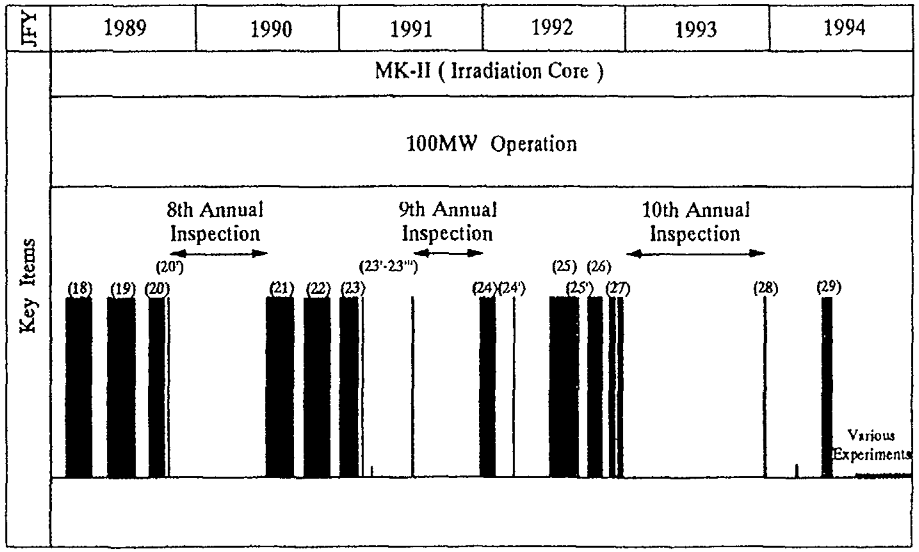


Fig. 2.1 Core Configuration for the 29th Duty Cycle





Operation Time ; 51,100 hr
Accumlated
Thermal Power ; 4,156,111 MWh
(as of March 31,1995)

Fig. 2.2 Operating History of Experimental Fast Reactor JOYO

3. Prototype FBR, Monju

3.1 Construction Schedule

The Monju site is located on the northside of the Tsuruga Peninsula in the central Japan, facing the Sea of Japan and is surrounded by mountains of approximately 300-700m high. Since the plant is located inside the Wakasa Bay Quasi-national Park, its construction works have been carried out with special attention to the environment.

Major milestones of the construction schedule (shown in Fig. 3.1) are as follows;

Oct. 1985	Start of Construction
Apr. 1987	Completion of Construction of the Reactor Containment Vessel
Oct. 1988	Installation of Reactor Vessel
Apr. 1991	Completion of Construction

3.2 Present Status

The equipment installation was finished and the construction was completed in Apr. 1991. The function tests were performed from May 1991 to Dec.1992. After that, the start-up tests have been performed. The reactor power has been increased gradually from Feb.1995. Generating electricity and connecting to the grid will be started in Jul. 1995.

3.3 Function Tests

The schedule in Fig. 3.2 shows the general outline of function tests.

Function tests were carried out from May 1991 to December 1992.

Function tests were conducted to confirm the function and performance of the plant systems, following various tests and inspections during fabrication and installation of the components in Monju.

The tests are divided into three phases:

1) Testing the fuel handling system and control rod drive mechanism in air at room temperature prior to sodium charging.

2) Tests in argon gas before loading sodium into the systems. Argon is used for preheating and heatup.

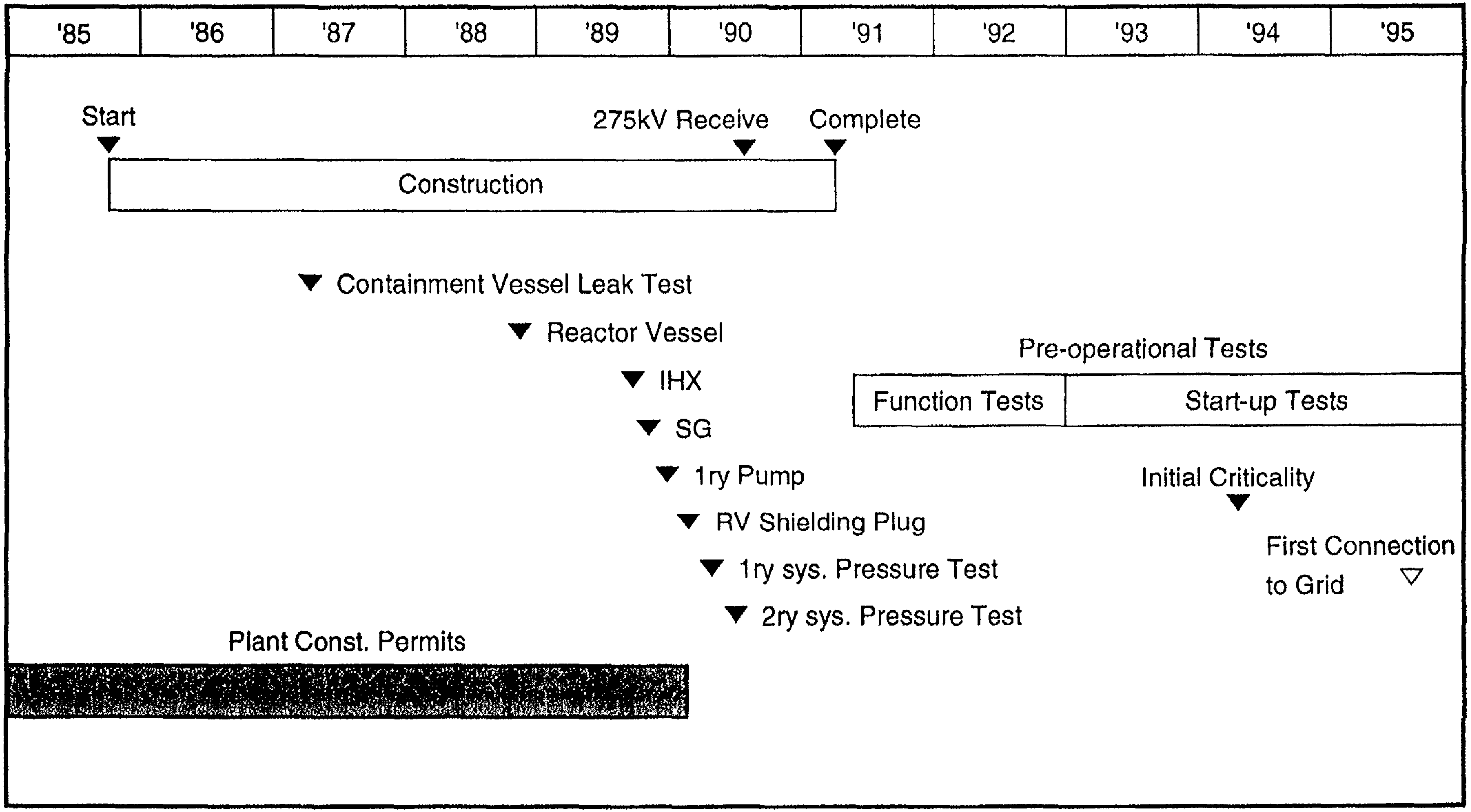


Fig. 3.1 Monju Construction & Tests Schedule

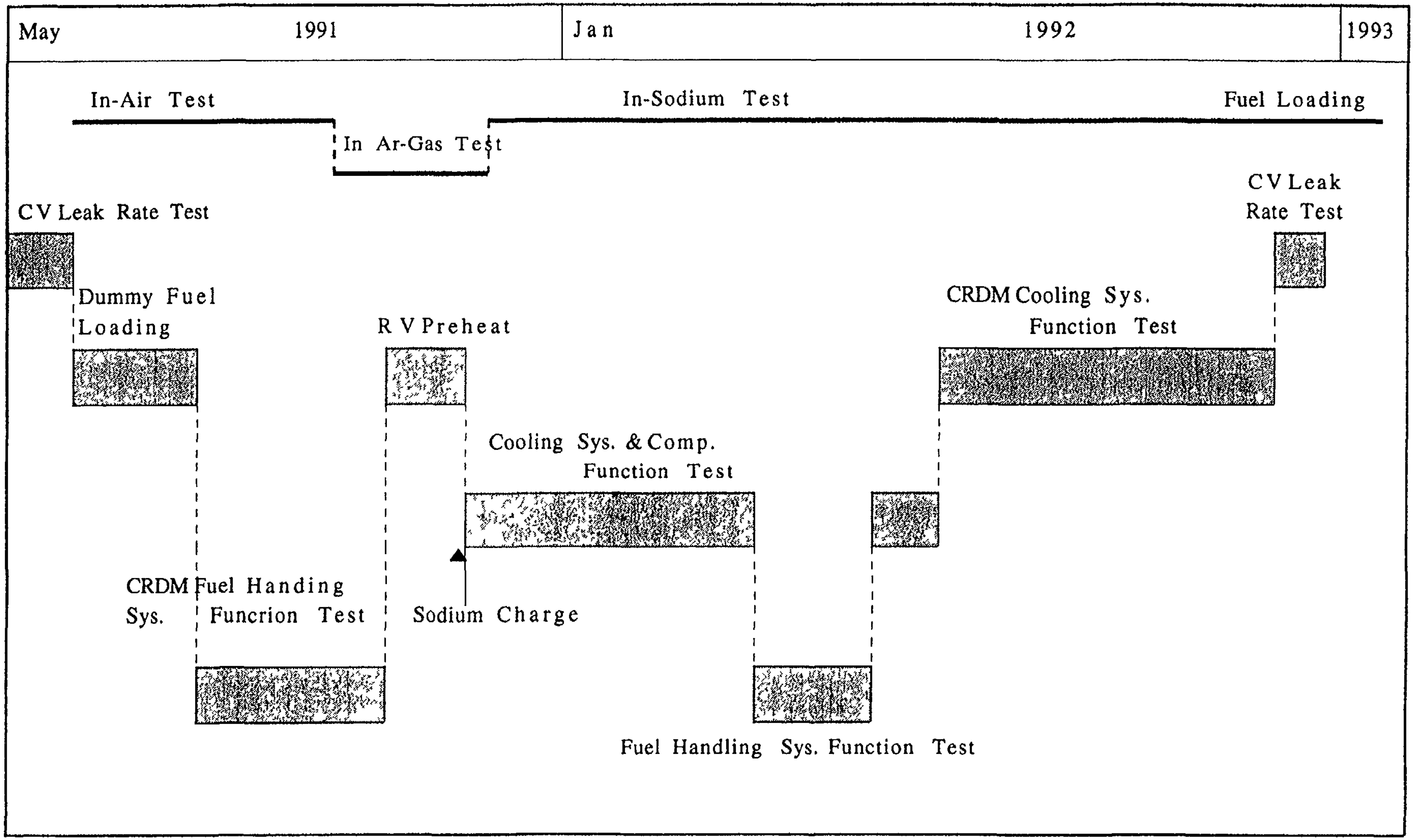


Fig. 3.2 Monju Function Tests Schedule

3) Testing the cooling systems, control systems, fuel handling systems etc. after sodium loading.

There were about 300 system function tests, of which 240 are specific to the FBR. Included in these are:

1) The configuration of a dummy core

2) Confirmation of the operation of fuel handling components in gases and sodium

3) Confirmation of the movement of in-service inspection and pre-service inspection equipment

4) Preheating of the system components and sodium charging

5) Leak rate measurement of the reactor containment vessel after loading the sodium

3.4 Start-up Tests

The schedule in Fig. 3.3 shows the general outline of the start-up tests.

The purpose of the start-up tests is to confirm and to evaluate the performance of the core, the plant systems and the components.

The start-up tests consist of criticality test, reactor physics test, power-up test, etc. They are somewhat similar to the LWR's ones.

First of all, the pre-performance test of plant system was conducted to examine plant characteristics under high temperature condition without nuclear heating.

Initial criticality was achieved with 168 fuel assemblies in Apr. 1994.

After the criticality test, the reactor physics test was performed and core reactivity worth, core reaction rate distribution , core flow rate distribution, etc. were measured.

The nuclear heating test has started in Feb.1995 and the reactor power is going to increase gradually. Monju will be connected to the grid in Jul.1995. After that, power-up test will be started. In the power-up test, plant characteristics under power operation and transient condition will be confirmed.

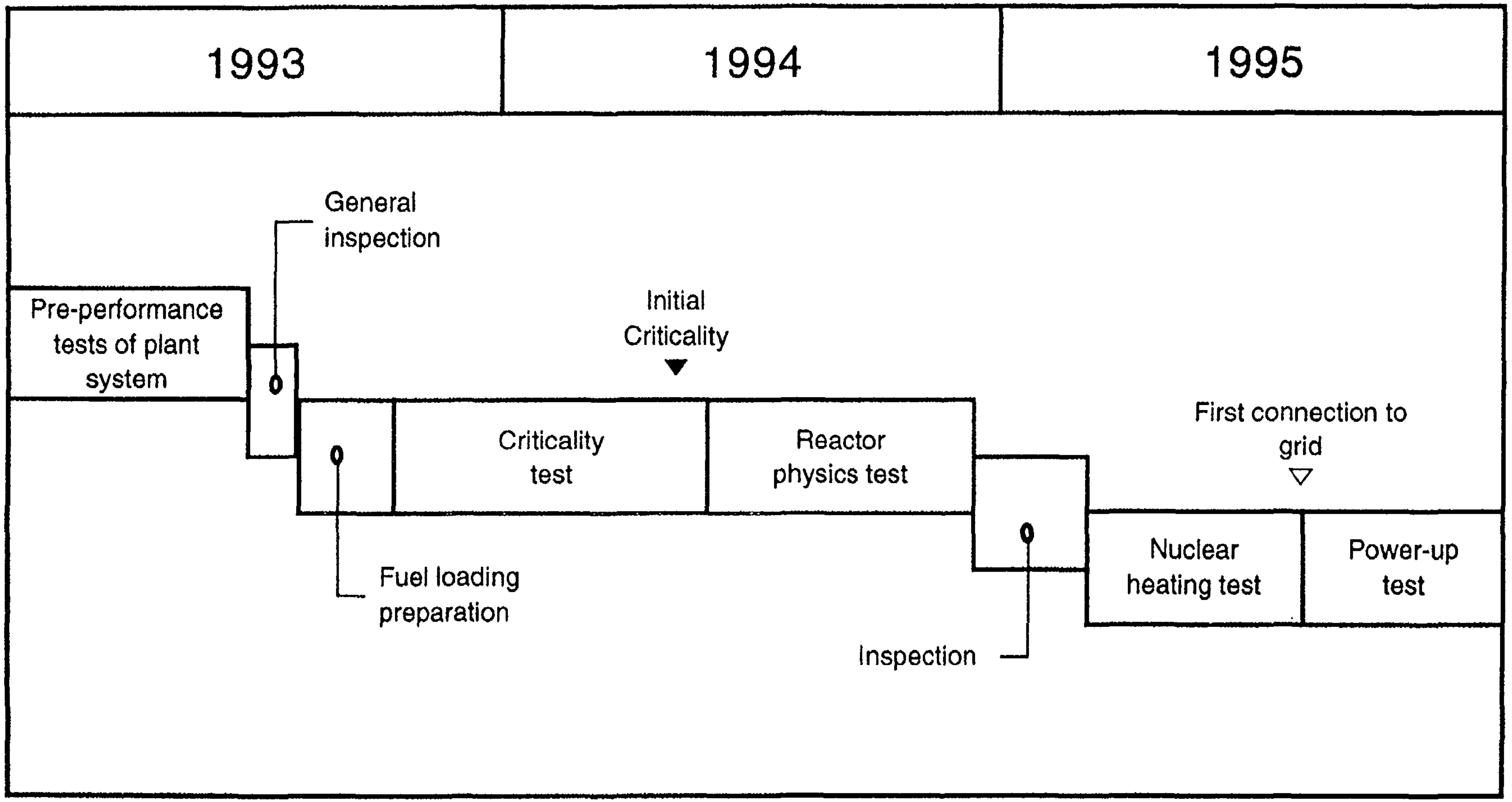


Fig. 3.3. Main schedule of Monju start-up tests

Table 3-1 **Principal Monju Plant Design Characteristics**

Reactor Type		Sodium cooled FBR, loop-type
Thermal Power		714 MW
Gross Electrical Power		280 MW
Core	Equivalent Diameter	Approx. 180 cm
	Height	93 cm
	Volume	2340 lits.
Fuel		PuO_2 - UO_2
Pu Enrichment	(Pu fissile)	(Inner core/outer core)
	Initial Core	15/20%
	Equilibrium Core	16/21%
Fuel Inventory	Core(U+Pu metal)	5.9 ton
	Blanket (U metal)	17.5 ton
Average Burn-up		Approx. 80,000 MWD/T
Cladding Material		SUS316
Cladding Outside Diameter/Thickness		6.5/0.47 mm
Permissible Cladding Temperature (middle of thickness)		675 °C
Power Density		275KW/lit
Blanket Thickness		Upper 30 cm
		Lower 35 cm
		Radial 30 cm
Breeding Ratio		Approx. 1.2
Reactor inlet/outlet Sodium Temperature		397/529 °C
Secondary Sodium Temperature (IHX inlet/outlet)		325/505 °C
Reactor Vessel (height/diameter)		18/7m
Number of Loops		3
Pump Position (Primary and Secondary Loop)		Cold Leg
Type of Steam Generator		Helical Coil, once-thorough Unit Type
Steam Pressure (Turbine Inlet)		127 kg/cm^2 g
Steam Temperature (Turbine Inlet)		483 °C
Refueling System		Single Rotating Plug with Fixed Arm FHM
Refueling Interval		Approx. 6 Months

4. DFBR and PNC's Design Study

4.1 Overview

The Japan Atomic Energy Commission (JAEC) promulgated Japanese "Long-term Program for Development and Utilization of Nuclear Energy" in June 1994. In the program, it was concluded that the research and development for demonstration FBRs (DFBRs) should be done with the cooperation of governmental and private sectors, and that utilities should play the major role in design, construction and operation of the DFBR aiming at the commercialization by the year about 2030 through construction of two DFBRs with a step-by-step improvement of technologies and economics.

The start of construction of DFBR-1 is expected in the early 2000's in the program.

4.2 Design Study of DFBR

The Japan Atomic Power Company (JAPC) has conducted for the past several years conceptual design studies of the Demonstration Fast Breeder Reactor (DFBR) in accordance with the basic policy of the Federation of Electric Power Companies (FEPC) and has confirmed the feasibility of top entry loop type reactor.

Based on the result of these design studies, FEPC decided in January 1994 to start the construction of the DFBR plant at the beginning of 2000's, and the major specifications.

Currently a new phase of design study is in progress for the three years from FY1994 to FY1996, based on the major specification and those design study. These studies have been focused on core safety enhancement, increase of design margins for the structural tolerance of the containment facilities, feasibility and licensability of seismic isolation plant for introduction to the DFBR. Through these studies, the goal to make the design of the whole DFBR plant harmonious with both safety and economic viability, and to prove preparation for the basic design.

4.3 PNC's Design Study

PNC has defined 10 key technical issues to be attained for commercialization of FBRs. In 1988, PNC started plant design study applying key technologies such as reactor vessel head access concept and performed plant construction cost evaluation.

PNC conducted design study on a 600MWe-size plant for 1990 - 1991. In 1992, JAPC and PNC discussed on the mutual design results to improve the demonstration reactor design study.

Design study on a 1300MWe-size plant featured with passive safety using nitride fuel is underway at PNC.

PNC is also studying new recycling system using new type fuel (nitride, metal) and minor actinide, and is considering the necessities of test equipment and facilities for the development of this system.

5. Reactor Physics

5.1 Development of Analytical Method

In order to treat the Hex-Z geometry of FBR cores more accurately, effort to develop a neutron transport code based on an improved nodal method was continued. In the previous version of the nodal code, the radial transverse leakage on node boundaries was assumed to be distributed uniformly, which generates some truncation errors. This year, a new treatment for the radial transverse leakage was introduced to the code by adopting a second order polynomial expansion of the flux at the node vertex point. A benchmark test of an FBR core showed the new nodal method can predict the keff within errors of 0.02%dk/k, on the other hand, the previous treatment has errors of 0.1%dk/k.

An investigation to improve prediction capability of Doppler reactivity experiment in critical assemblies was performed. The calculation/experiment (C/E) values were extremely underestimated by the conventional method with a cell group constant set and isolated lump model. A new calculation with ultra-fine energy group cross-sections was applied to the Doppler experiments, and resulted in the improvement of C/E values by 13%. From the present survey, it is found quite significant for the Doppler reactivity analysis to take into account the interaction effect between Doppler samples and core fuel.

5.2 Cross-Section Adjustment

For FBR cores, it is substantial to improve the prediction accuracy of the burnup characteristics such as burnup reactivity loss, number density change of nuclides and breeding ratio. In this context, a code system to calculate the cross-section sensitivity of the burnup characteristics has been developed and completed based on a generalized perturbation theory successfully.

Doppler reactivity is the other important parameter in FBR cores because it assures the intrinsic safety characteristics of the core. The present cross-section adjustment method cannot treat the Doppler reactivity which is dominated by resonance-peak broadening of cross-sections. We have launched a new study to extend the applicability of the cross-section adjustment and design accuracy evaluation system to the Doppler effect. The basic method to evaluate sensitivity of self-shielding factors has been successfully derived from a generalized perturbation theory, and a prototype system to calculate the Doppler sensitivity is now under verification.

5.3 Critical Experiment

A critical experiment for nitride fuel FBR cores is in progress at Fast Critical Assembly (FCA) as a cooperative study between Japan Atomic Energy Research Institute and PNC. The first stage of the experiment, which consists of reaction rate and sample reactivity measurement in small area of nitride fuel, was completed on June 1994. The preliminary analysis showed the general prediction accuracy of the nitride fuel core seemed to be equivalent to that of the conventional oxide fuel cores. The second phase of the nitride fuel core experiment with enlarged region of nitride fuel is planned in 1996.

5.4 Shielding Experiment and Analysis

Post-analysis of Japanese-American Shielding Program for Experimental Research (JASPER) was continued. The Flux Monitor Experiment and the Special Materials Experiment were evaluated in detail. Analytical results of JASPER were effectively utilized in the Demonstration FBR shielding design review, which was conducted in the joint study of Japan Atomic Power Company (JAPC).

As development of analytical method for shielding characteristics, a three-dimensional discrete ordinate transport code TORT was broadly tested and verified to possess excellent performance for analysis of complicated configurations.

6. Systems and Components

6.1 Reactor Shutdown System

Research on the self-actuated shut-down system (SASS) by using a curie point magnet is in progress. A system performance test on SASS by use of Joyo is under planning based on various out pile characteristic tests conducted in air and sodium environment.

6.2 Process Instrumentation

A development of ISI test equipment for Monju was almost completed in O-arai engineering center(OEC). Mock-up test rigs and ISI test equipments at OEC were disassembled and transferred to Monju from OEC.

A development of the visualization technique of heat transfer tube inner side by using Laser is in progress.

6.3 Steam Generator

PNC precedes a conceptual design study for a future FBR plant having a steam generator in the primary heat transport system. The studies on a double-wall tube(DWT) steam generator have been in progress regarding structural integrity and thermal hydraulic performance of the DWT steam generator.

The 1MW scale DWT steam generator test model has been operated since 1991. So far, good thermal hydraulic performance was validated. Four of 10 heat transfer tubes were plugged to obtain thermal hydraulic performance at higher water side mass velocity.

6.4 Sodium Component

An in-sodium performance test of a magnetic flux concentration type electro magnetic pump (EMP) was completed. An evaluation of the EMP is underway.

6.5 Thermal Hydraulics

A verification of plant dynamics simulation code (Super-COPD) by using Monju functional tests was carried out and the accuracy of the models was validated.

Predictions for the Monju start-up test were carried out beforehand and the verification by using the test is planned.

7. Fuels and Materials

7.1 Fuel Fabrication

The PFPF (Plutonium Fuel Production Facility) equipped with automated and remote handling fuel production systems started fabricating Joyo and Monju fuels from October 1988.

7.2 Fuel Pin Performance

Fuel pin performance codes for analyzing the steady state and transient conditions have been improved since 1984, with the data of PNC/DOE operational reliability testing program in EBR-II, etc. The modification of codes for annular fuel is emphasized. A three dimensional FEM code for an analysis of defected fuel pin behavior is modifying by PIE results of Run Beyond Cladding Breach (RBCB) pin. Development of nitride fel fuel performance code is also in progress.

7.3 Core Material

PNC1520 advanced austenitic steel was developed for high burnup FBR fuels. Excellent performance is demonstrated by the out-of-reacter and material irradiation testing programs. Design base standard of PNC1520 is completed and evaluation study is in progress to apply Monju and DFBR cores.

7.4 Irradiation Experiments

1) Joyo

Fuel pin irradiation is in progress with advanced austenitic cladding fuel pins and high strength ferritic cladding fuel pins. Large diameter annular pellet fuel pins for Monju high burnup and DFBR cores are included in the test.

The fuel subassembly with CEA austenitic stainless steel cladding tubes has also been irradiated since 1988 and reached 125GWd/t. The second power-to-melt test was conducted to analyze local fuel melting below 20% of fresh pins.

2) Foreign Reactors

Phase-II program of PNC/DOE collaborative operational reliability testing in EBR-II has been successfully continued and will be completed by end of 1995. Large amount of data base on TOP (transient over power) and RBCB (run beyond cladding breach) tests is summerized for typical MOX fuel pins.

The PIE of fuel subassemblies of PNC316 steel and PNC1520 advanced austenitic stainless steel irradiated in FFTF is in progress.

7.5 Development of Advanced Fuels

Feasibility study of advanced fuels (nitride, metal, carbide) has been conducted since 1986. PNC/JAERI irradiation test of nitride and carbide pins has been conducted in JOYO since 1994.

Mixed carbide fuel pins were also irradiated using the thermal reactor JRR-2 and JMTR of JAERI.

7.6 Post Irradiation Examination

Construction of PIE facility of Fuels Monitoring Facility (FMF-2) adjacent to existing FMF-1 at OEC is in progress to handle the large fuel assemblies irradiated in FBR Monju and ATR Fugen. Hot operation of FMF-2 will start from late 1996.

8. Structural Design and Materials

8.1 Development of Structural Design Method

1) FINAS nonlinear structural analysis program

Effort to enhance the capavility of the general purpose nonlinear structural analysis program FINAS is continued, particularly with respect to adoptive mesh generation based on r-method and h-method. FINAS was mounted as the solver in CAE systems such as CADAS, ATLAS, FEMAP and so on. FINAS is currently used by many engineeres at about 40 sites including fabricators and universities. The latest version, V12.0, was translated into English.

Personal computer version of FINAS was developed.

2) Improvement of Elevated Temperature Structural Design Guide

The following rules are investigated to improve and extend the current Elevated Temperature Structural Design Guide.

i) Creep-fatigue design methods based on elastic analysis
A new cree-fatigue design method, which is based on the concept of a generalized elastic follow-up model, is being developed. The elastic follow-up equations to predict strain magnification and creep relaxation for structural discontinuities are established.

ii) Design rules for weldment
A new design approach, taking into account the metallurgical and geometrical dicontinuities inherent in weldment, is being pursued.

iii) Strain limit criteria
A ratchetting criteria for multiaxial stress state, which are not provided explicitly in the design guide, was developed, and is being examined the applicability of the criteria to general components by FE Analyses.

8.2 Structural Test and Evaluation

Structural tests are being performed to improve strength prediction methods, to evaluate the adequacy of elevated temperature design rules, and also to verify advanced nonlinear structural analysis methods.

1) Thermal creep-fatigue test with small sodium loops (SPTT and STST)

A thermal creep-fatigue test, whose specimen with welded joints was made of 9Cr-1Mo steel, was finished, and then the test facility (STST) was modified to complete a new test condition of a new test model "a cylindrical shell with cross-section gradually step-changing".

2) Thermal transient tests in large sodium loop (TTS)

A thermal transient test of a vessel with fillets model is now under way. A new test model, made of FBR grade 316 stainless steel, was fabricated.

3) Distortion tests of fuel sub-assembly duct (CMDT, WFT)

SCFT and BHAT equipment for Plastic Buckling tests were modified and new equipment, CMDT and WFT, will be constructed to perform a distortion test of ducts of a fuel sub-assembly. Then a new R&D, which is to investigate the distortion behaviors of internal structures, will be started in near future.

8.3 Seismic Test and Analysis

1) A conceptual study on the vertical seismic isolation system for FBR components is underway. A series of shaking table tests and analytical works are included to assess the feasibility of the system.
2) Seismic analysis method development and verification on FBR core is underway in the framework of "IAEA/IWGFR Coordinated Research Program on Intercomparison of LMFBR Seismic Analysis Codes".

8.4 Fracture Mechanics and Structural Integrity Assesment

Both deterministic and probabilistic fracture mechanics methodologies are being developed for the integrity assesment of flawed or cracked structures.

Computer codes developed at PNC, CANIS-J for calculation of fracture mechanics parameters, and CANIS-G for simplified crack propagation tests of a cylinder with circumferential and axial flaws and with a cylinder with an axial temperature gradient conduted by the Air-cooling Thermal Transient Test Facility(ATTF) are continued.

8.5 Structural Material Tests and Evaluation

Structural material tests in air, in sodium, in water/steam, and under post-irradiation condition have been conducted to revise the Monju Material Strength Standard and to prepare a new version for DFBR.

The test program in air and in sodium environment is called "Capella" program and the step-1 program (1985-1987) , the step-2 program (1988-1990) and the step-3 (1991-1993) were already completed. The step-4 are currently underway with emphasis on long-term extrapolation.

The post-neutron irradiation tests are underway within the scope of neutron irradiation program "Spica".

1) Tests in Air

The present Capella step-4 program includes following subjects;

- Validation of long-term exptrapolation of a new criterion for creep-fatigue failure strength of weldment, inelastic constitutive equations on new materials(modified 9Cr-1Mo steel and FBR grade 316 stainless steel)
- Improvement of LBB evaluation method for FBR plant.
- Development of the material strength standard for modified (Cr-1Mo steel and FBR grade 316 stainless steel

  A new concept creep-fatigue failure criterion was proposed using secondary creep basis ductility exhaustion for stainless steels. Applicability of the present criterion to modified 9Cr-1Mo steel was validated by the mechanical and metallurgical studies. The equations of cyclic plasticity of FBR grade 316 stainless steel were revised based on the data-base consisted with low cyclic fatigue and tensile test data.

2) Test in Sodium and Water/steam

Mechanical strength (tensile,creep fatigue,creep fatigue) tests on 316FR (nitrogen controlled) in sodium are still continued in the program to evaluate the carbon and nitrogen transfer effects. Low-cycle fatigue and creep fatigue tests on modified 9Cr-1Mo steel in water/steam environment were conducted.

3) Tests in Irradiation Environments

Surveillance tests for the Class 1 components of Joyo were condunted to confirm the integrity of the reactor by evaluating irradiation effects of the same materials.

The test data were used for the planning of Joyo operating program.

Tests for the Class 1 components of Monju to evaluate irradiation effects on the machanical properties up to the end of design life and to evaluate irradiation effects on the Material Strength Standard for Monju are also in progress.

Both forged and rolled SUS304 steels were irradiated in Joyo using SMIR (Structural Material Irradiation Rig).

Another test for DFBR was conducted to clear the relationship between creep rupture strength and metallurgical variables such as chemical composition, grain size and production process.

Several post irradiation material tests and in-pile creep tests on FBR grade 316 stainless steel were continued in Joyo and JMTR (Japan Material Test Reactor of JAERI) in accordance with the R&D program Spica step-2.

4) Data Processing System

Material data are complied using specific data coding sheets, and the data inputs to the computer data processing system SMAT are still continued.

Entry data in SMAT are currently more than 12,000 data points on 11 different kinds of mechanical tests (including tensile, low cycle fatigue, creep) for 10 kinds of FBR structural steels.

9. Safety

9.1 Safety Evaluation for Normal and Abnormal Events

Safety evaluation studies have been conducted for confirming the physical phenomena and integrity of the reactor fuel elements and the structures in the primary system during the normal operation, scram transients, and the early stage of postulated accidents. Recently major emphasis has been placed on an evaluation of passive safety features such as decay heat removal by natural circulation.

The potential to remove decay heat by natural circulation is one of the important safety features of LMFBRs. To enhance the decay heat removal capability under the case of loss of all electrical power, experimental studies on natural convection in a reactor vessel are in progress using sodium as a working fluid.

An integral sodium experiment has been carried out with a partial core model composed of seven subassemblies, inter-wrapper gaps, an upper plenum and a dipped cooler. A series of tests is underway focusing on inter wrapper flow, core-plenum interaction under conditions of decay heat removal by natural circulation.

Studies on development, validation and application of thermal-hydraulics analysis codes are in progress for safety analysis and evaluation of normal, upset and accident conditions. Regarding the whole plant level simulation, the SSC code has been developed and validated. Currently, the pre-test analysis of Monju natural circulation test is being performed for integrated validation of the code. Also, the SSC code is applied to the performance evaluation of inherent and passive safety features to establish an advanced FBR concept from the viewpoint of the safety.

Lineup of three dimensional thermal hydraulics simulation is a finite difference method code AQUA; SPLASH based on Arbitrary-Eulerian-Lagrangian finite element method; and direct simulation code DINUS-3. The AQUA code is being applied to natural circulation and thermal stratification analysis. The SPLASH mostly simulates the free surface phenomena. The DINUS-3 code simulates thermal striping phenomena. Validations of these codes are almost completed and practical problems are currently solved.

Thermal-hydraulics in fuel subassemblies are analyzed using subchannel code ASFRE for single phase flow and SABENA for two-phase flow. The ASFRE code is currently under validation study using PNC's sodium experiment data mentioned above. Although there is little activity regarding the SABENA code, it is ready for utilization in the safety evaluation of demonstration FBR. What is noted in this reporting period is a new finite element code SPIRAL for thermal-hydraulic simulation inside the fuel bundle surrounded by extremely complicated geometry. The new code will be used for more microscopic temperature and flow field analysis in the fuel assembly that will be required in the local fault evaluation.

9.2 Degraded Core Research

In the context of international cooperation in this area under a framework of IWGFR, the technical committee meeting on "Material-Coolant Interactions and Material Movement and Relocation in Liquid Metal Fast Reactors" was held at the O-arai Engineering Center, PNC in June 1994. The meeting successfully covered and compiled the current states of the art of broad research areas in the degraded core safety. The proceedings to the meeting was distributed to the IWGFR member states.

The degraded core research at PNC addresses: the fuel failure propagation during local-fault accidents, and physical phenomena during core disruptive accidents (CDAs).

The local fault studies include out-of-pile experiments on local coolant blockages by porous media with water to confirm fuel pin integrity through analyses using a detailed sub-channel code, ASFRE. A new series of sodium experiments is also planned. To focus on an intra-subassembly failure propagation behavior and to establish a termination scenario, a synthesis study is in progress by reviewing and interpreting the past in- and out-of-pile experimental data base, such as SCARABEE, MOL 7C, SLSF and TREAT.

The current out-of-pile experimental program at PNC consists of various simulant melt experiments using the MELT-II facility. A series of experiments to investigate the erosion behavior of solid structures by a high-temperature molten jet was completed. Experiments to study thermal interactions between a molten jet and coolant are in progress. A low-temperature series of tests with Woods metal and water has been completed. From the experiments, four distinct modes of interaction behaviors were observed. Conditions necessary for energetic interactions were identified, based on consideration of the minimum film boiling temperature. From extrapolation to reactor materials, it is predicted that such conditions are unlikely to be met. High-temperature experiments with alumina and sodium are planned for the next few years.

On the in-pile experiments jointly conducted with French CEA, a synthesis work was completed for the CABRI-2 program. In the on-going CABRI-FAST program through 1995, slow and fast transient tests have been conducted, mainly with high burnup, annular fuel pins. Program planning for a next joint in-pile test program is being positively discussed between PNC and CEA, as well.

The CDA analysis code development and validation studies have continued extensively: SAS4A for CDA's initiating phase and SIMMER-III for the transition phase. For SAS4A, under collaboration with KfK and CEA, a reference code version REF-94 has been developed for common use in the future. A special single channel code PAPAS-2S, which models detailed fuel pin mechanics, is also being elaborated through CABRI analyses. The fluid-dynamics system in SIMMER-III has approached its completion and the coupling with the neutronics is underway. A code assessment program is jointly participated by KfK and CEA. Also underway is the development of a new interface code SAME-II that couples SAS4A and SIMMER-III. The system of new-generation codes are to be applied to CDA studies of D-FBR and future fast reactors.

Finally, a long-term research program, which has been undertaken over the last several years at PNC, must be mentioned here. This program aims at identifying long-term research needs of integral in-pile safety experiments, which are essential to further advance safety technologies towards FBR commercialization in the next century. The areas of primary interest include: eliminating recriticality concerns during CDAs, demonstrating advanced fuel design and types, and establishing local-fault scenarios. A feasibility study is in progress to define required features and performance of the in-pile test facility, and to establish a basic design concept of a reference core.

9.3 Plant Accident Research

FBR plant accident research consists of two major activities. One is a study on a non-radiological sodium fire caused by sodium leakage from the intermediate heat transport system (IHTS), and the other is a study on the radiological source term, with emphasis being placed on quantifying various mitigation factors of fission product (FP) release and transport from failed fuels to the environment. The latter study also includes an integrity assessment study of the reactor containment with respect to FP leakage during a severe accident.

In the sodium fire study, experiments have been continuously conducted using the SAPFIRE facility. Emphasis is placed on studies of large leak rate fires and related multidimensional spatial effects in the combustion process. Obtained data are used for validation of the three-dimensional computer code SOLFAS.

In the source term study, a new experimental rig has been constructed for investigation of FP release behavior from high temperature fuel under an accidental condition. In this rig, the fuel can be heated up to 3,000°C by induction heating. Various performance tests are in progress to validate experimental procedures and measurements. Experiments using irradiated fuels of Joyo are planned in near future. The FP bubble behavior in sodium and the FP release behavior from sodium pool surface are studied with the SABER and the START rigs, respectively. The TRACER code is under development to describe various FP behavior in an in-vessel sodium system. For an ex-vessel study, the hydrogen burn experiments are continued in atmospheric conditions containing sodium aerosols or mists. The CONTAIN-LMR code is continuously improved for analyses of overall severe accident progression in a containment.

9.4 Steam Generator Safety Research

Current steam generator (SG) safety researches consist of two major activities. The one is for improvement of the sodium-water reaction evaluation method for the large-scale demonstration FBR plant. For this purpose, an overheating failure propagation process is studied in detail because this may potentially determine the design base leak (DBL) of future plant SG's. A series of simulation experiments in the TRUST rig has been conducted to examine structural performance of high-temperature tube under accident conditions. Obtained data are analyzed using the computer codes, FINAS and AUTODYN. Since the TRUST experiments simulate reaction heat by means of induction heating, large scale sodium-water reaction experiments are planned in the modified SWAT-3 facility. Effects such as sodium flow and tube cooling by water system blowdown will be simulated in the large scale experiments.

Another study for the SG safety is aimed for the FBR design concept which eliminates the secondary sodium loop. To support this conceptual feasibility study, double-wall SG design is proposed and examined. Since a reliable leak detection system is essential for this design, detailed investigation of tube failure detectability is performed. A helium gas system in the gap of double-wall tube is used for detection. In case of an inner tube failure, steam in the helium system is detected. On the contrary, helium gas contents in sodium is detected in case of an outer tube failure. Both experimental studies and code development are in progress to validate this detection system.

9.5 Research on Probabilistic Safety Assessment

PNC has been performing the research on Probabilistic Safety Assessment (PSA) for more than ten years as part of the R&D of a fast reactor.

The purpose of this research is to construct probabilistic safety models for a typical loop-type FBR plant so that an overall safety assessment can be performed. It is expected that (1) a systematic evaluation on the plant safety is conducted based on the quantitative analysis, (2) the insights on measures to enhance system reliability and safety are provided, (3) the operation and maintenance procedures are established based on a risk-based consideration, and (4) useful information is given to the development of basic policy on safety design and evaluation of a large LMFBR.

PNC has been improving the systems analysis code network which is able to perform a level-1 PSA. Recent efforts have focused on improvement in a phased-mission analysis program based on Monte Carlo method, development of dynamic event sequence analysis technique dealing with human intervention, and development of a Living PSA System (LIPSAS). The LIPSAS has been installed at the site of Monju plant to examine the applicability of the system to safety management of a real plant. Furthermore, PNC has been developing a new software SAGE (Severe Accident Guidance using Expertise) which is able to quantify the consequence of core damage quickly with the expert system that models accident phenomena based on the experience of full scope PSA.

Efforts are being made to develop a new relational data base system of LMFBR component reliability data on an engineering work station. The system is based on CREDO (Centralized Reliability Data Organization), a cooperative project between PNC and the USDOE, which ended in 1992. As part of the data analysis reliability parameters were quantitatively estimated for sodium mechanical pumps, i.e., failure rates, probability of common cause failures and repairability. Additionally, risk-related data on energy production systems such as solar photo voltaic energy system and LMFBR nuclear fuel cycle have been collected.

Scoping study on plant-shutdown PSA has been completed and dominant core damage sequences have been identified. A detailed shutdown PSA is underway. Preliminary application of PSA to large LMFBR is underway to provide basic information in developing safety design and evaluation policy. The fifteen initiating event categories were identified and the associated event trees were developed. The reactor shutdown system and the decay heat removal system were modeled based on fault trees. A phased-mission analysis approach was used to evaluate the decay heat removal system. Also the study on classification of safety function importance for LMFBRs is ongoing. Reliability goal of PS (prevention system) and MS (mitigation system) has been examined based on the insights obtained from the PSA application.

Level-2 PSA tasks (consequence analysis) are underway. The current effort includes an analysis of dominant core damage sequences such as an unprotected loss-of-flow (ULOF) accident in a large LMFBR. For the in-vessel physical process, a preliminary analysis with SAS3D and SIMMER II is continued to identify generic core behaviors especially from the viewpoint of energetics potential. For the ex-vessel physical process parametric analysis of key event sequences in a containment is continued to comprehend the sensitivity of phenomenological and design-related parameters such as containment volume, design pressure, leakage rate, amount of ejected sodium, etc.

10. Fuel Cycle

10.1 Mox Fuel Fabrication

1) Construction and Fuel Fabrication

R&D on fabrication of uranium-plutonium mixed oxide (MOX) fuel has been carried out since 1965 at the Plutonium Fuel Development Facility (PFDF) in Tokai works of PNC.

The Plutonium Fuel Fabrication Facility (PFFF), which started operation in 1972, has two fuel fabrication lines for Advanced Thermal Reactor (ATR) (10 ton MOX/year) and FBR (1 ton MOX/year). It has supplied the fuel necessary for the operation of ATR Fugen and FBR Joyo.

In parallel with the construction of Monju, construction of the Plutonium Fuel Production Facility (PFPF) (FBR line; 5 ton MOX/year) started in July 1982. It was designed to develop fuel fabrication technologies as well as to fabricate fuels for Monju and Joyo. The construction was completed in October 1987. After testing operation, production of Joyo fuel started in October 1988 as the first production campaign at PFPF. Production of the initial core fuel of Monju started in October 1989 and was completed in January 1994 at PFPF.

The PFPF is currently fabricating fuels for Joyo.

Also in 1994, PNC started construction of a new line (10 ton MOX/year) at PFPF so as to produce fuels for ATR Fugen, because of PFPF ATR line's becoming superannuated.

PNC is planning to construct another ATR line at PFPF so as to produce fuels for the ATR demonstration reactor.

The present Japanese suppliers of uranium fuel and PNC will also cooperate to make increased use of PFPF to manufacture MOX fuel, for large scale demonstration of plutonium use in LWRs in Japan.

About 140 tons of MOX fuel have been fabricated by the end of March 1995.

2) R&D on MOX Fuel Fabrication

Remotely controlled operation technology is one of the most important key element to achieve a large scale production of MOX fuel.

PFPF equipments including material transfer system are designed and manufactured so as to realize the fully automated operation.

Through the operation of PFPF FBR line so far, PNC has been accumulating experience for it.

10.2 Plutonium and Uranium Conversion

PNC developed a co-conversion technology utilizing the microwave heating direct denitration process (MH method) which converts plutonium nitrate and uranyl nitrate solution to MOX powder. Compared with the conventional method, it is a simple process and generates less liquid waste.

The Plutonium Conversion Development Facility (PCDF) (conversion capacity: 10 kg MOX/d), designed for demonstration of the co-conversion technology by MH method, was completed in February 1983. By the end of March 1994, it produced about

9.1 ton of MOX powder containing about 3.7 ton of plutonium. The converted MOX powder were transported to PFFF and PFPF, in addition to about 1.8 ton of MOX powder processed at another small scale facility, and are being used for fabrication of MOX fuel for Fugen, Joyo and Monju.

Since reprocessed uranium through reprocessing of spent fuel has generally higher U235 concentration compared to natural uranium, our country has decided to try to use it as LWR fuel by re-enriching and mixing it with other enriched uranium and by mixing with plutonium as fuels for ATR, etc..

Reconstruction of pilot scale reprocessed uranium conversion facility was completed in June, 1994.

Conversion test for reprocessed uranium was started in August, 1994.

11. FBR Fuel Recycling

In the area of FBR fuel reprocessing, PNC has developed process and equipment with remote handling technique, through large scale cold mock-up tests at the three Engineering Demonstration Facilities (EDFs) and laboratory scale hot tests at the Chemical Processing Facility (CPF) in Tokai Works, on the basis of accumulated experience in the Tokai reprocessing plant for LWR fuels.

The construction of Recycle Equipment Test Facility (RETF), where advanced process and equipment tests in engineering scale under hot conditions are conducted in order to enhance the technology and economical efficiency, is initiated in January, 1995.

Almost all of R&D activities are oriented towards the RETF project.

11.1 Process Research and Development

1) Head End process

In order to remove the hexagonal wrapper tube efficiently prior to fuel chopping, a disassembly system with CO_2 laser has been developed and tested. A reference cutting scenario has been established through tests with dummy fuel assemblies. A prototype test equipment of geometrically safe continuous rotary dissolover was fabricated and tested.

2) Chemical Separation Process

Major effort of solvent extraction contactor development is paid on centrifugal contactor. The design of the prototype contactor for RETF has been completed successfully.

In order to eliminate the generation of secondary salt-bearing waste in the purex process, studies and tests on solvent cleanup with salt-free reagents and electro-reoxidation process for Pu have been continued.

3) Common Technology

To establish remote maintenance concept with rack system, key technologies such as bilateral servo manipulator (BSM), roll-in type rack, remote connector bank, and remote sampling system have been developed.

Materials of process equipment and advanced analytical system are also under development.

4) Hot Tests at CPF

Irradiated fuel from Joyo, Phenix, and DFR with burnup up to 99,800 MWD/T have been used for hot tests at CPF.

Through these hot tests, information of dissolution characteristics dependent on many factors and nuclides behavior in the solvent extraction process have been obtained. In addition, PNC has developed the Np co-extraction process with U and Pu to enhance non-proliferation aspect.

11.2 Plant Design of Recycling Facilities

1) Recycle Equipment Test Facility (RETF)

Verification of high availability and economical prospects of FBR fuel reprocessing are essential for deployment of FBR and its fuel cycle. In order to accomplish them at future pilot plant, hot engineering demonstrations of important process and equipment are necessary in advance. From this viewpoint, PNC has planned the RETF project to provide a test bed for advanced equipment and process.

RETF features a large remote cell which accommodates both head-end and chemical process equipment test areas. Most of the chemical processes will be mounted on the racks installed along either cell wall. The maintenance of these chemical process equipment as well mechanical components will be conducted by using overhead crane and bilateral servo-manipulator (BSM).

RETF is scheduled to start hot tests in the late 2000.

It is expected that important data and experience for designing of future FBR fuel reprocessing plants will be compiled through the RETF operation.

2) FBR Fuel Recycling Pilot Plant

PNC has a plan to construct the FBR Fuel Recycling Pilot Plant to verify the whole plant system and it will be the first step for achieving the cost requirement.

The operation will be initiated in the middle of 2010s.

LIQUID METAL-COOLED FAST REACTOR B-350 AS A HEAT SOURCE FOR SEAWATER DESALINATION AND ELECTRICITY PRODUCTION

V.L. LEVITIN
Mangushlak Energy Group of Enterprises,
Kazakhstan

Abstract

The BN-350 nuclear power plant has been used for seawater desalination and electricity production for 23 years, the longest of any commercial liquid metal fast reactor (LMFR). During that time much has been learned about successful LMFR operation and design. The present paper describes some important design features of the BN-350 NPP and presents some performance results from the whole operational period.

1. INTRODUCTION AND SUMMARY

The Mangushlak peninsula in the Republic of Kazakhstan is located far from communication routes. The area has rich natural resources but with limited sources of fresh water and energy. For the development of this land plenty and reliable supply of fresh and drinking water and energy were urgently needed. The problems were solved in 1973 when the nuclear power plant with liquid metal-cooled fast reactor, BN-350, had been constructed on the shore of the Caspian Sea (on the Mangushlak peninsula). The BN-350 is the first and the only nuclear power plant in the world producing heat for seawater desalination. The BN-350 reactor has an electric power output of up to 130 MW(e); in addition 80000 tonnes of desalinated water are produced per day for the city of Aktau during more than 20 years. A new big modern city with gardens and parks has been developed in the desert.

Due to changes in the former USSR, limited information on the BN-350 reactor, which now belongs to the Republic of Kazakhstan, has been available to the international community in last years. This paper intends to partly fill this gap.

2. REACTOR PLANT

Reactor: The loop type design concept of the BN 350 reactor plant (Figs. 1,2,3) provides for a separate arrangement of the equipment. There are 6 loops, each with a pump and intermediate heat exchanger (IHX), two valves to isolate the loop, and a check valve in the cold leg (between the pump and the isolation valve). The primary sodium pipes are contained in cells which are steel lined and have a gap between the lining and the walls to permit cooling. In the event of radioactivity in the cells the inward ventilation will be blocked and the outward ventilation routed directly to filters and then released to the atmosphere. The sodium is circulated in the primary and secondary circuits by centrifugal pumps (Fig.4). The primary sodium pumps have two operational speeds, 100% and 25%. Only a maximum of 5 loops may operate together. All 6 pumps operating at full speed would give unacceptable pressure at the bottom of the vessel. When all loops are available, one is kept in a hot-standby condition. The argon pressure in the gas cavities of the reactor and the primary pumps is 0.19 MPa (1.9 bar). Each primary loop is provided with two valves at the pressure and suction lines. The sodium temperatures at the reactor inlet and outlet are rather low at full power (520 MWth): 290 and 440°C respectively.

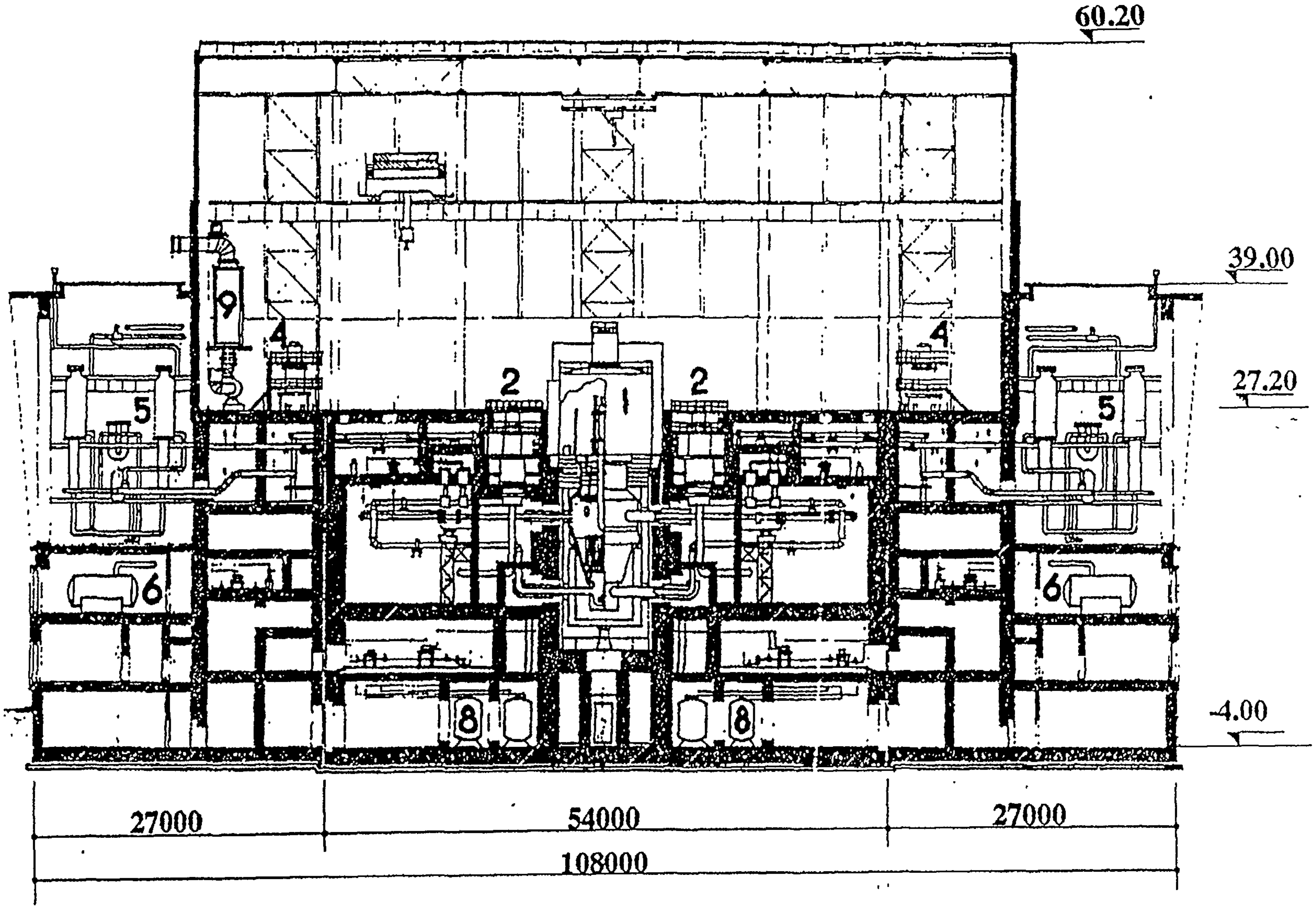


FIG. 1. Vertical longitudial-section.

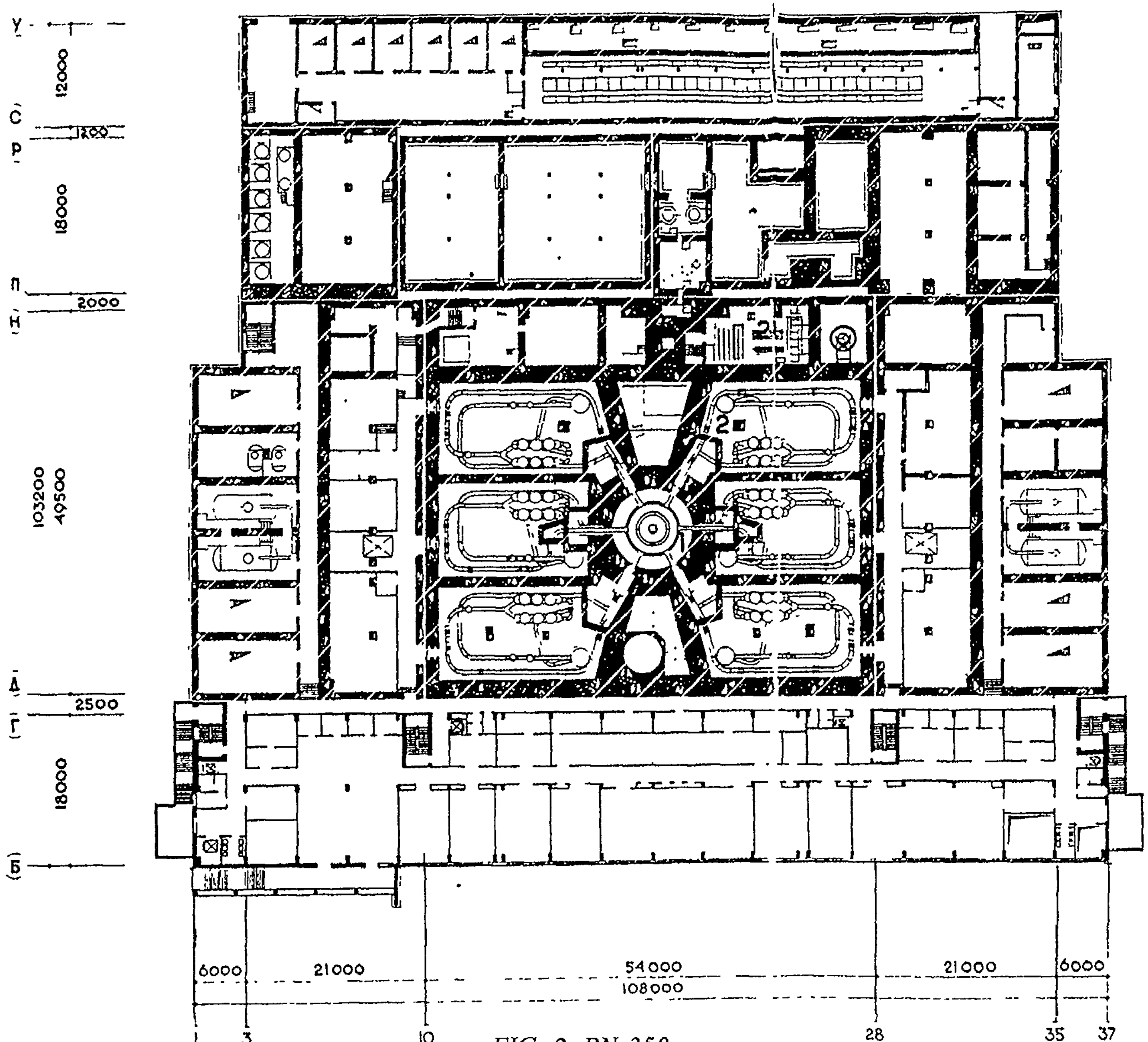


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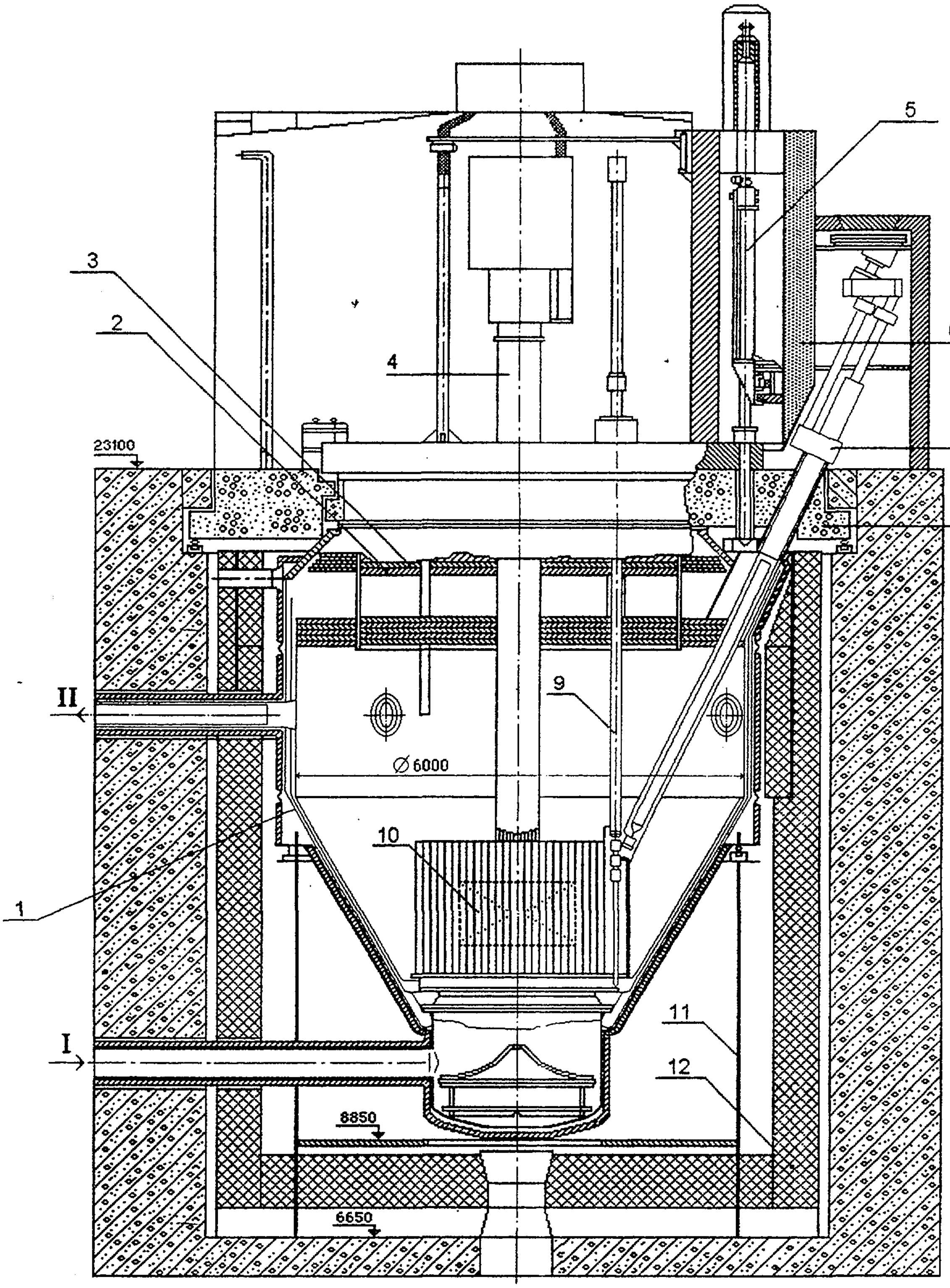
FIG. 2. BN-350.



1-shell
2-primary plug
3-secondary rotary plug
4-central tower with protection control mechanism
5-assembly handling mechanism
6-handling box
7-handling elevator
8-fixed upper box
9-handling mechanism
10-core
11-reactor rest
12-lateral protection
I-sodium inlet
II-sodium outlet

FIG. 3. The BN-350 reactor cut.

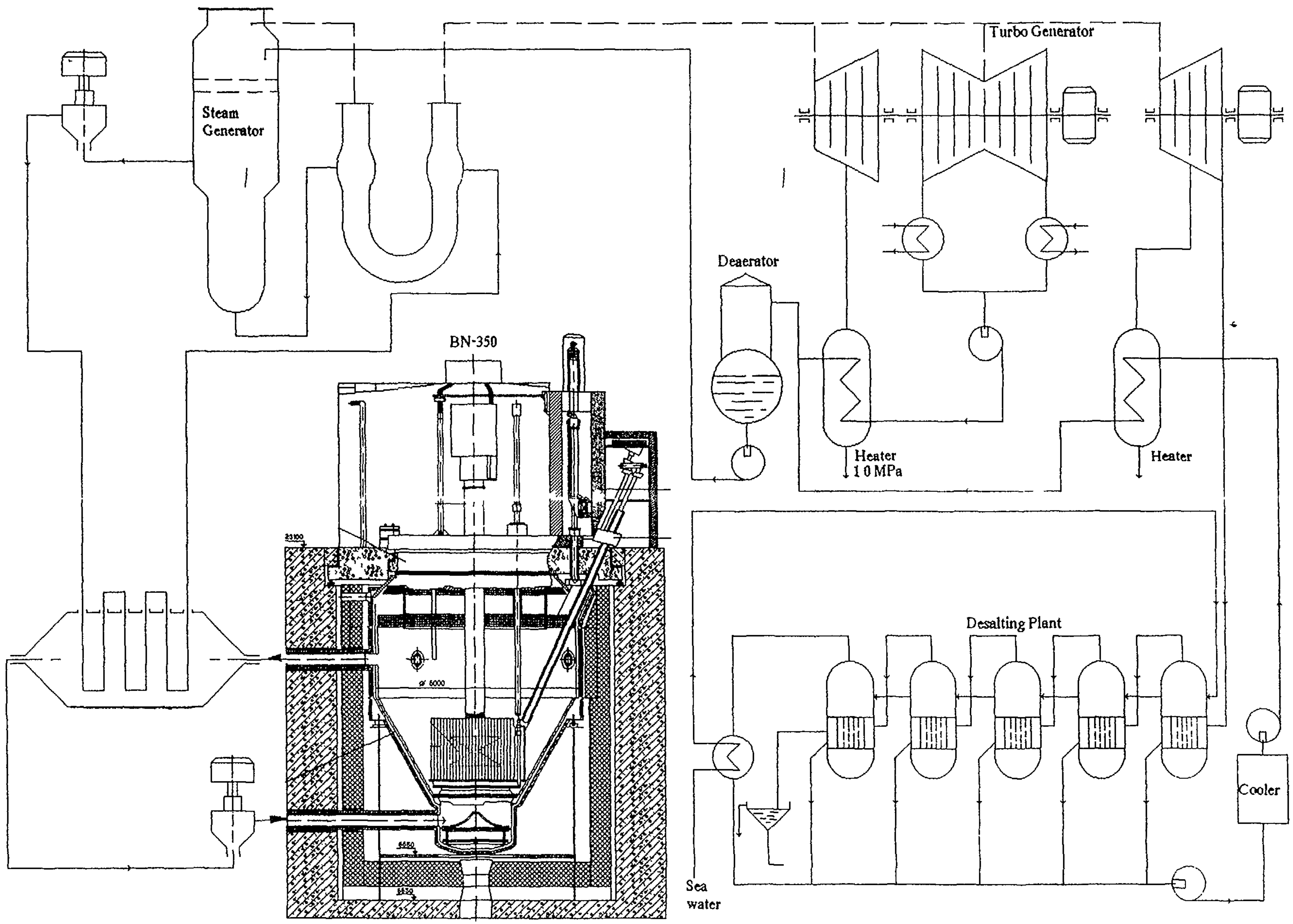


FIG. 4. Flow dıagram of the nuclear desaltıng complex

The fuel subassemblies of the core and of the blanket are placed in the discharge header mounted on the pressure chamber of the reactor vessel (fig.5). The core loading of 220 subassemblies contains fuel elements with UO_2 fuel and a blanket material. More recently some MOX experimental fuel elements were also loaded into the core. There are 12 positions in the core for the rods of the control and safety system. Of the 12 control rods 2 are automatic control rods, 6 are reactivity compensators, 3 are scram rods and 1 is a temperature effect compensator. The core is surrounded by the blanket, containing depleted uranium dioxide. The axial blanket is mounted in the subassemblies of the core (Fig. 6) and the radial blanket is formed by fuel subassemblies containing blanket material. Above the core is a central column containing the control rod drives and thermocouples. The column is only as wide as necessary for the absorber rod systems; it does not cover the full width of the active core zone.

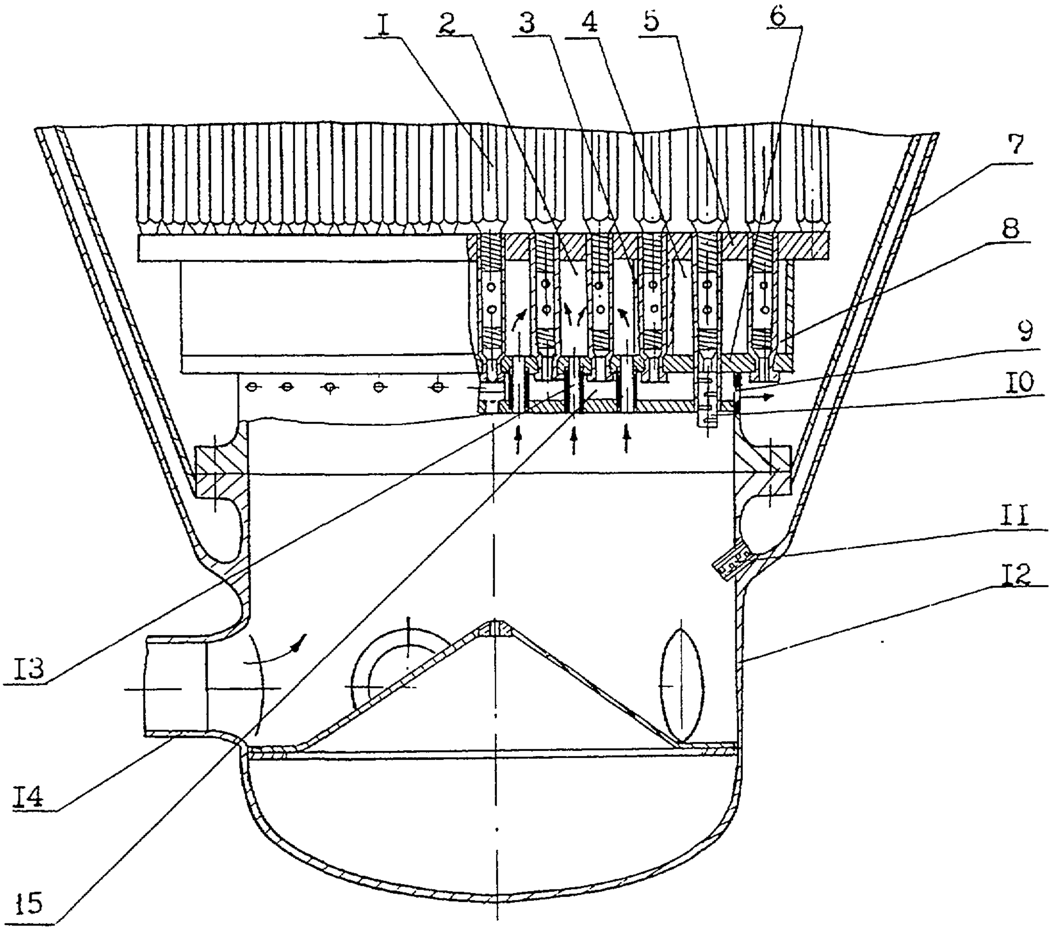


FIG. 5. Elevation of the BN-350 diagrid and pressure plenum.

1-fuel subassembly; 2-high pressure plenum; 3-partition; 4-low pressure plenum; 5-upper plate of the pressure plenum; 6-lower plate of the pressure plenum; 7-reactor vessel; 8-collector of sodium flowing from the lower volute spring sealing of the fuel subassemblies; 9-collector draining orifices; 10-low pressure plenum throttle; 11-throttle for the reactor vessel cooling sodium flow; 12-reactor inlet sodium collector; 13-high pressure plenum inlet orifices; 14-high pressure pipe socket; 15-low pressure collector.

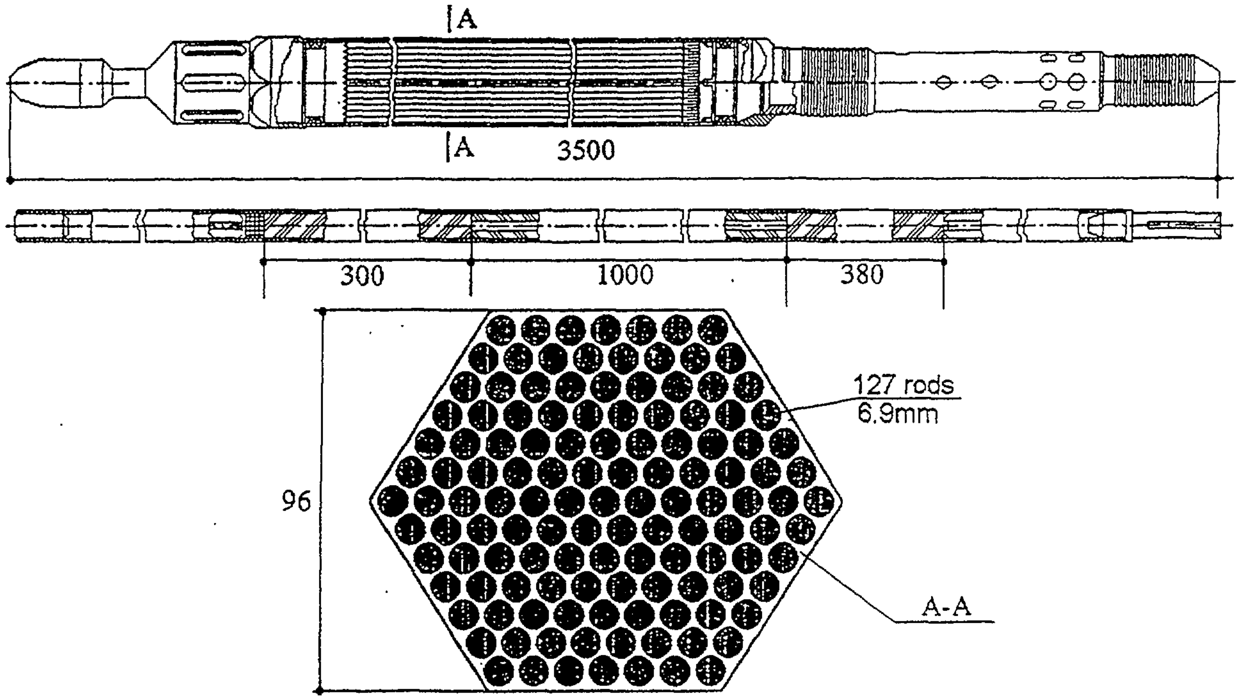


FIG. 6. Fuel assembly and rod.

The reactor has a vessel of a variable diameter (the maximum diameter is 6000 mm) made of austenitic stainless steel. The lower part of the vessel forms a pressure chamber for sodium which flows into the chamber through the piping from the pumps (Fig. 3). The main reactor vessel is supported from below at the top of the conical section about half-way up the vessel. The diagrid rests on a ring rising from the bottom of the conical section. The main reactor vessel is surrounded by a guard vessel and there are leak jackets on the primary sodium pipes from the vessel up to the isolation valves. The vessel is placed in an iron oxide covered box for shielding. The sodium is heated while passing in the upward direction through the core, the blanket and the storage and flows through the upper mixing chamber of the vessel and the piping to the heat exchangers. The inner surface of the vessel and the outlet nozzles are provided with screens reducing temperature stresses arising from fast changes of the coolant temperature. The vessel is cooled by a cold sodium flowing from the pressure chamber in the space between the vessel walls and a thermal shield. The biological shielding outside the reactor is made of an iron ore concentrate, graphite, steel and concrete.

In the bottom of the reactor vessel is a flow defector which also serves to protect the vessel bottom and disperse debris in the event of a severe accident.

Flow through the hot pool is well mixed with the exception of the cooler region above the shielding at the top of the pool. A leakage flow from the cold pool is supplied to keep the walls of the reactor vessel cool.

Nuclear heat transfer to desalination plant: The nuclear power plant supplies steam to three turbogenerators and to a desalination plant (80 000 tonnes of desalinated water per day). To transfer the heat from the reactor to the intermediate heat exchangers six primary parallel sodium loops and six independent secondary sodium loops are provided. The turbine exhaust steam under the pressure of 0.6 MPa (6 bar) is supplied to the desalination plant, and the condensate produced with the temperature of about 100°C flows to a heater and a deaerator and is pumped by feedwater pumps to the natural circulation steam generators (Fig. 4).

Fuel handling: Refuelling is carried out by two direct lift charge machines, the small rotating plug, and angle ramps to introduce/remove fuel in the vessel. One machine is for fresh fuel, the other for spent fuel with 38 positions. Spent fuel is taken to the washing station where it is cleaned (water-vapour and nitrogen) and monitored for gas leaks. The assemblies are stored in a pond; leakers are first sealed in cans. The fuel is handled in a gas which can cool up to a maximum power of 5 kW. Two rotating plugs are mounted on the head of the vessel providing for positioning of the fuelling device above the given subassembly in the core or in the blanket. The rotating plugs also serve as an upper biological shielding. Hydraulic seals with an eutectic alloy are used to provide an airtight fit to the rotating plugs.

The charging and discharging of fuel subassemblies is performed by the following mechanisms (Fig. 7, 8): (1) a refuelling machine mounted on the small rotating plug which reshuffles the fuel subassemblies inside the reactor; (2) charging-discharging elevators transferring the fuel subassemblies from the reactor to the transfer box and back, and (3) a transfer mechanism for fuel subassemblies mounted in an airtight box transferring the spent fuel subassemblies from the reactor to an outside storage and fresh fuel subassemblies to the reactor.

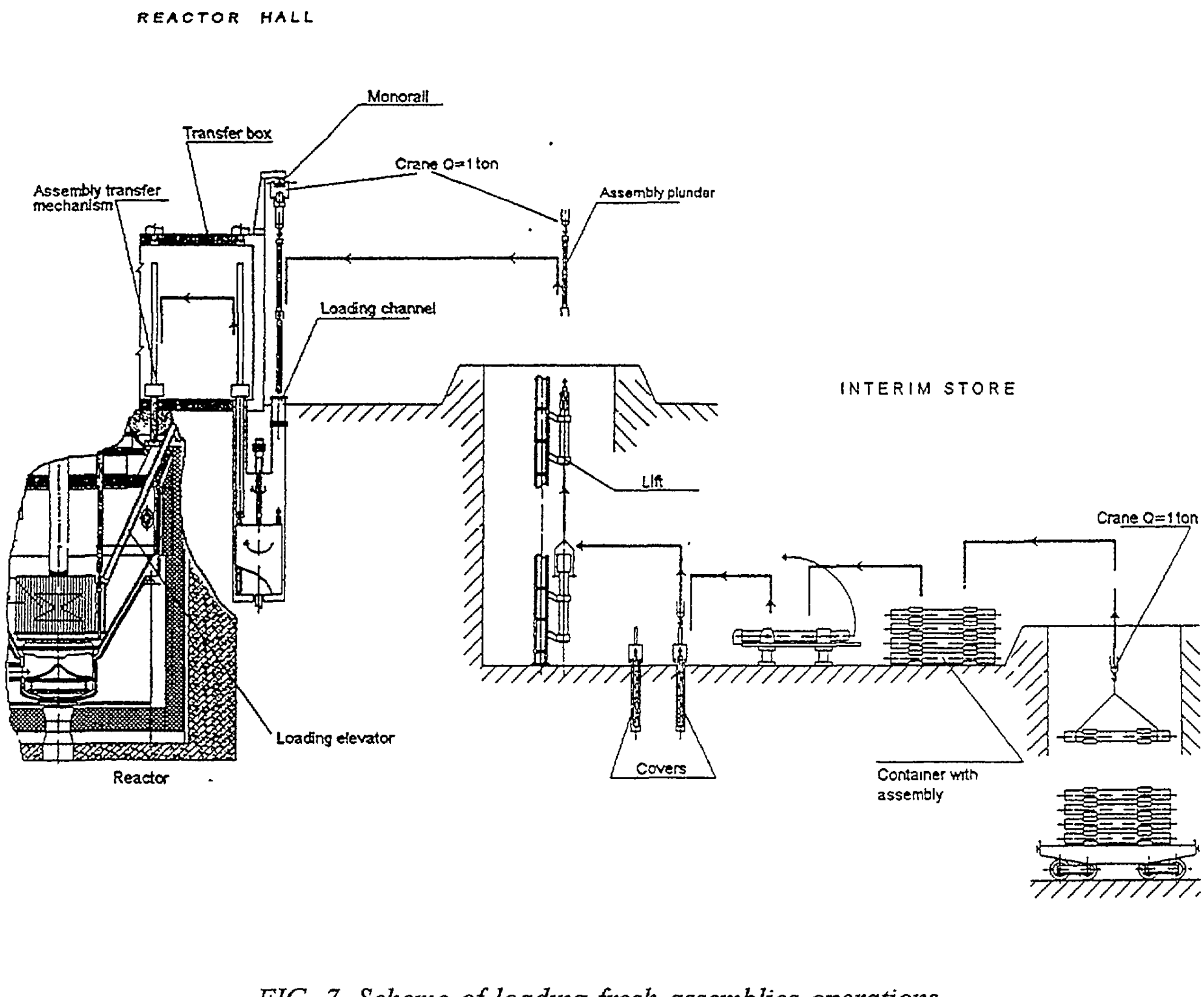


FIG. 7. Scheme of loading fresh assemblies operations.

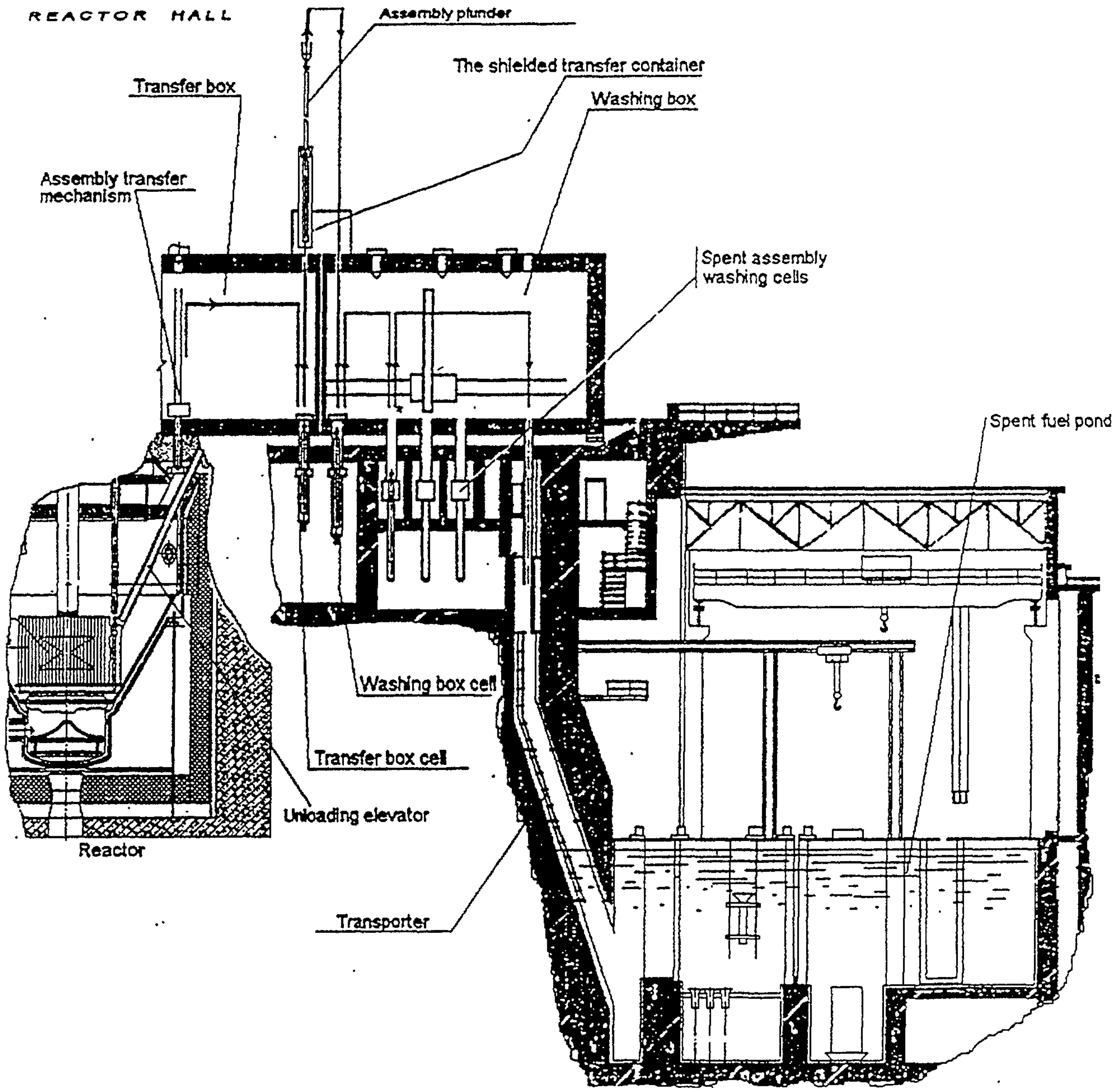


FIG. 8. Scheme of unloading spent assemblies operations.

Special safety features of the BN-350 are: (1) steel enclosure around the rotating shield plug and the fuel transfer equipment, and (2) negative sodium void coefficient.

Secondary circuit: Each circuit comprises an intermediate heat exchanger (IHX), secondary sodium pump, steam generator (SG) and sodium pipework. The BN-350 plant uses two types of steam generators. Four loops have modular type steam generators of a unique design. Two loops have multi-modular Czechoslovakian steam generators. These have 64 modules containing sodium and water/steam tubes. To detect leaks, hydrogen monitoring in the sodium and gas space is provided. The modules are contained in a box with air vents at the bottom and top (with a stack) to allow air cooling by natural convection when no feedwater is available. This provides an alternative decay heat removal route. The secondary pumps are similar to the primary pumps in design. The tops of the IHXs are provided with special seals so that the primary and secondary pipework are in separate cells within the reactor building.

3. OPERATING EXPERIENCE

Plant history (Fig. 9).

1965-71	Construction period
1972	First criticality of the reactor
1973	Power startup of the reactor
End of 1973	Two major steam generator defects (leakages at bottom end of evaporator tubes and subsequent sodium/water reaction)
Feb. 1975	Another steam generator leakage; sodium/water reaction (800 kg) and sodium fire for 2 h, no explosion effects. Cause of leakage: cracks on tube surfaces had not been detected during inspection
1973-1975	Operation at power levels up to 300 MW(th), study of physical and thermal characteristics, inspection and overhaul of 5 (out of 6) steam generators
1975	Increase from 37 to 50% nominal power
From April 1976 - Jan. 1989	Operation at 650-700 MW(th) power level for seawater desalination and electrical power generation.

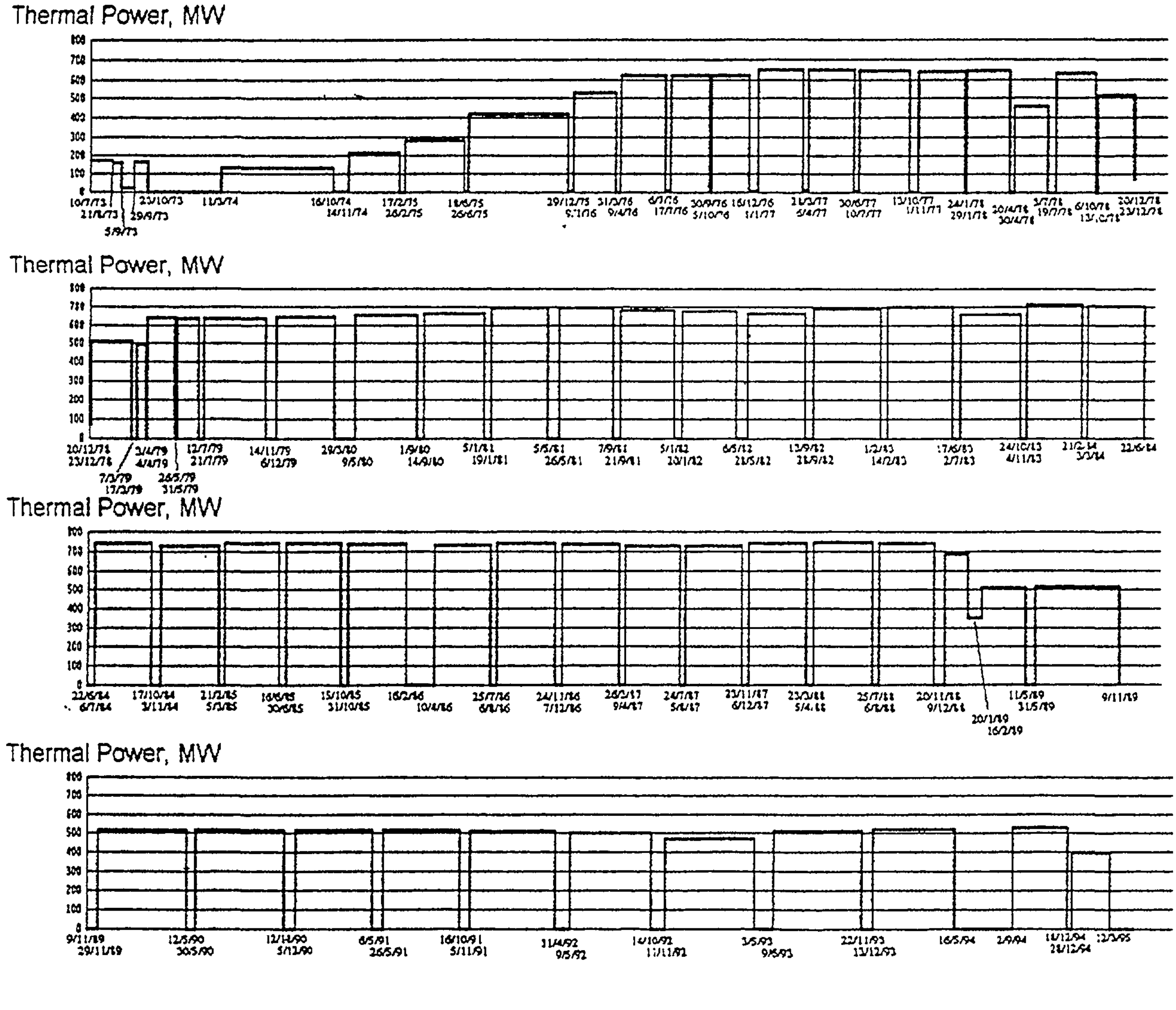


FIG. 9. Reactor BN-350 work history.

From Jan. 1989 (to present time)	Operation at 500-520 MW(t) power level for seawater desalination and electrical power generation.

Reactor core: Basic difficulties in achievement of reliable and effective operation of the BN-350 core were connected with some irradiation impacts on strucutural material properties, selection of non-optimal materials, which under the irradiation of about 50-60 displacement per atom (dpa) suffer a considerable swelling and fragility that cannot allow continued normal operation of hexagonal tubes of sub-assemblies, cover tubes of absorbing assemblies and their elements. The results of a special research programme on irradiated materials allowed consequently to improve the BN-350 core and increase maximum fuel burn-up from 5% of heavy atoms up to 7%, then 8.3% and now about 12%. Simultaneously, the life time of absorbing assemblies was increased from 60 to 450 effective days.

Reactor and sodium circuit: After solving the steam generator problem in 1974-76, two problems that were connected with sodium vapour and aerosol deposition on cold surfaces were revealed at the following stages of reactor operation.

The first of them was an increased resistance to rotation of the rotating plugs in the process of the core refuelling. It was found that the design temperature conditions of the refuelling process were not optimum: at a prescribed coolant temperature of 170°C the required heating up of the rotating plugs was not ensured. For handling of this problem the temperature was increased to 250°C, and now there are no problems with rotation of the plugs.

The second problem was plugging of gas lines with the sodium vapour condensate. For handling this problem some of the gas pipelines were replaced by larger diameter pipes and provided with heating and draining systems.

With respect to secondary-circuit operation problems arose by the steam generator leaks in 1973-1975. For ensuring the required coolant quality after the leaks prolonged operation of coolant-purification systems was called for. A great quantity of impurities, mainly of sodium-water interaction products, had accumulated in cold traps so that the operating efficiency of the traps began to decrease. To restore purification system performance, regeneration work on four secondary circuit cold traps and coolant preparation system cold traps was carried out.

The regeneration method essentially consisted of conversion of sodium oxide to hydroxide which has of melting point not exceeding 400°C. At higher temperatures hydroxide is easily removed from the trap into the sodium transportation vessel by pressurizing of the trap.

After the regeneration, the efficiency of the cold traps was practically completely restored, the volume of impurities retained in the trap did not exceed 10% of the working volume of the settling tank.

All sodium-circuit equipment expect for the above one has been operating satisfactory. One should especially note adequate operation of the sodium valves.

The design of the check valves has been refined, and the valves in the primary loops have been replaced by advanced ones. An algorithm by which reactor power is automatically reduced by a preset value without reactor shut-down at switching-off of one of the primary pumps has been introduced.

The main gate valves have adequate operating characteristics; they assure such a tightness level of a disconnected loop that operations on withdrawal of removable parts of the main pumps and check valves can be carried out without reactor shut-down.

4. PROLONGATION OF OPERATION

The BN-350 reactor was designed for an estimated 20 year life-time. The reactor has now been operating for 23 years. Considering the importance of the nuclear power plant for the region, both as a source of desalinated water and electricity, it is the intention to extend the operation of the reactor for as long as it is safe, efficient and economical.

The originally assumed life-time of most components of the reactor equipment has been exceeded already. Presently, assessments of the residual life-time for major components and the reactor as a whole are being carried out. It should to taken into account that the reactor design was begun in 1960, and it was carried out in accordance with general-purpose industry requirements and old nuclear safety standards. The present, stringent safety requirements now call for a number of special arrangements. A comprehensive study of the design, with regard to its compliance with contemporary standards, is now being carried out, and on the basis of this analysis a decision will be taken about actual perspectives for the future operation of the reactor.

A BRIEF SUMMARY OF THE KOREAN LIQUID METAL REACTOR R&D PROGRAMME

MANN CHO
Korea Atomic Energy Research Institute,
Taejon, Republic of Korea

Abstract

This paper summarizes the status of the Korean nuclear power grid and gives the background for the LMR R&D works in Korea.

The main objectives of the R&D program and a tentative schedule with 5 progressing stages for the development of KALIMER(Korea Advanced LIquid MEtal Reactor) are given and the current LMR experiences are briefly described.

1. Background

A. Nuclear Plant Program in KOREA

A-1. Current Status

- PWR 9 Units in Operation (GR-1,2,3,4 YG-1,2,3 UG-1,2)
  3 Units under Construction (YG-4 UG-3,4)
- PHWR 1 Unit in Operation (WS 1)
  3 Units in Construction (WS 2, 3, 4)
- Total Nuclear Capacity by Present : 8,616 MWe
  Total Electricity Generated in 1993 : 56.5TWh (43% of Total Electricity)

A-2. Nuclear Future Plan

- By 2006, 23 Nuclear Plants (for 20,016 MWe)
  . 18 PWRs will be operated (6 new plants to be ordered)
  . 5 PHWRs will be operated (1 new plant to be ordered)
- From 2007, Korean Next Generation Plant will be deployed
  . By 2001, the Technology Development will be completed
  . The Reference of This Plant is System 80+
- Long Term Plan
  . By 2030, 24 New Nuclear Plants will be needed
  . By 2011, A Demo./Prototype LMR will be constructed
  . To 2025, The Main Nuclear Plant will be PWR
  . From 2025, Commercial LMR will be deployed

B. Needs of an LMR Program

In the 21st century, it is expected that mankind will continue in population and economic growth and continue to provide measures to conserve the environment. In order to minimize a serious problem with the environment due to the green house effect, acid rain and the destruction of ozone layer, the necessity of nuclear power should be largely emphasized as an energy resource. Many developing countries will begin to introduce nuclear power plants due to their economic growth. Also in the developed countries, it will be more difficult to employ well-trained operators due to their social infra-structure. Therefore, the establishment of an inherently safe nuclear power plant will be indispensable in the 21st century, and it is expected that LMRs will have the potential capability to enhance safety and to resolve the TRU reduction and spent fuel storage problems through proliferation-resistent actinide recycling.

It has been anticipated that the worldwide development of LMR technology will be completed around 2010 when most demonstration reactors will have been finished in construction. The commercialization of LMRs has been expected to be realized around 2025 - 2030. In this context, the LMR R&D programs have been started in Korea at this time, and we have the objective to establish an advanced LMR design concept in order to contribute to a certain amount of worldwide LMR R&D works and ultimite LMR industry.

The technologically improvable areas for the development of economic and inherently safe LMRs could be: elimination of rotating plugs, simplification of in-vessel transfer machine, introduction of electro-magnetic pump, volume reduction of secondary sodium above the reactor level(against sodium fires), and introduction of seismic isolation. Recently, the above advanced concepts to be applicable to advanced LMRs are being developed, and Korea anticipates contributing to R&D works of those items.

2. LMR Development Plan

A. Objectives

- To develop the long-term nuclear industries for the LMR around 2025 being expected as the time of commercial introduction of LMRs.
- The completion of basic design of the Korean prototype LMR(KALIMER) by 2001
- The completion of KALIMER by 2011.

B. Prototype LMR Development Program Schedule

```
 '92              '97.7              '99.7                2001               2007                 2011
+-----------------+------------------+---------------------+-------------------+---------------------+
Basic Technology                                                                                First
Development        Concept Design                                                           Criticality
                                         Basic Design
                                                                 Detailed Design
                                                                                          Construction
```

C. Implementation Plan

- Development of LMR Basic Capabilities (1992. 9 - 1997. 7)
  . Development of basic technologies, specifically in :
    + Sodium handling and behaviors
    + Plant systems and components
    + Reactor systems
    + Passive safety and licensing
  . Establish top level specification
  . Establish LMR development strategy and plan
  . Establish international cooperation plans
- Conceptual Design (1997. 7 - 1999. 7)
  . Conceptual Design , including cost estimates
  . Define key systems and components
  . Define and initiate supporting development plan utilizing foreign facilities
  . Preliminary system design specification

- Basic Design (1999. 7 - 2001)
  . Define basic design including key components and detailed cost estimate
  . Continue supporting development plan
  . Prepare preliminary safety analysis report
  . Initiate site activities
  . Update strategy and define deployment plan
- Detailed Design (2002 - 2006)
  . Complete detailed design, including optimization and final cost estimate
  . Initiate fabrication planning and preparations
  . Complete safety analysis report and reviews and approvals
  . Obtain construction permit
- Construction and Startup (2007 - 2011)
  . Fabrication components and systems
  . Construction plant
  . Conduct startup tests

D. Expected Results
- Establishment of Commercial LMR Technology
- Contribution to Long-Term National Energy Security through Improved Utilization of Uranium Resources
- Improvement of Nuclear Safety by Development of Liquid Metal Reactor with Inherent Safety

3. Experiences on LMR Technology

A. '94 Activities
- Preliminary Work for Design Concept Establishment
  . Enhancement of Design Capability
  . KAERI-GE LMR Design Study
  . Future Detailed Program Setup
- Development of Principle Design Technology & Computation Codes
  . Core Design (Neutronics, T/H)
  . System Design (Thermo-fluid Engineering)
  . Mechanical Design (Mechanical/Structural & Seismic Analysis)
- Development of Na Experiment and Na Equipment
  . Experiment of Na loop & Measurement Techniques
  . Water Mock-up Experiment for Na T/H Characteristics
  . Na Equipment (EM Pump, Flowmeter/Levelmeter)
  . Experimental Na loop construction
  . SASS system
  . Na-material Compatability Analysis
- Development of Na Chemistry and Safety Measures
  . Na Fires
  . Na-Water, Na-Concrete Reaction
  . Na Purification

B. Joint Feasibility Study on the Introduction of a Commercial FBR Power Plant to Korea between Korea and France ('84 - '87)

- Set up the technological data basis for the planning of the FBR development or the FBR introduction strategy in Korea through the identification of the technological differences between FBR and PWR/PHWR systems.
- Prepare the implementation plan for the establishment of technological capability prior to the introduction of a commercial FBR power plant to Korea.

C. Joint Pre-design Concept Study on Korean Prototype LMR (KALIMER) with the comparison of GE's PRISMs and Japan's MDP concepts between KAERI and GE ('95 - '97)

4. Temporary Remarks on LMR in Korea

The commercial LMR project for 2025 should be started around 2015, and at least the prototype LMR should be operated for one equilibrium fuel cycle in order to get an invitation to bid from utilities at that time.

In the above framework, the strategy study has drawn up the LMR development directions and technology assurance strategy in order to faithfully attain the targets in the LMR long-term development plan, which are given as the following in summary:

1. Actinide recycle should be realized and the establishment of LMRs is required to continue nuclear power generation in the 21st century.

2. The worldwide commercialization of LMRs is anticipated to be realized around 2025 - 2030, and LMR development pursues the realization of commercial introduction of LMRs with the nuclear industry.

3. The construction of a prototype or demonstration type KALIMER(Korea Advanced LIquid MEtal Reactor) is essential to accomplish the goal of the LMR development. For the construction time, the year 2011 is suggested as established in the LMR long-term development plan.

4. The completion of basic design of KALIMER by 2001 should be kept as the mid target of the LMR long-term development plan.

5. The power output of KALIMER could be decided between 150 and 350 MWe, considering the consistency and the budget limit of the development program.

STATUS OF SODIUM COOLED FAST REACTOR DEVELOPMENT IN THE RUSSIAN FEDERATION

YU. BUKSHA
Institute of Physics and Power Engineering,
Obninsk, Russian Federation

Abstract

Considerable experience has been gained with the commercial fast reactor BN-600 in Russia. The present paper presents some performance results and gives an overview of the experience gained from the BN-600 reactor operation and maintenance in 1994. The status of R&D is stated in this paper. Special attention is given to plutonium and minor actinide burning.

1. THE STATUS OF NUCLEAR POWER IN 1994

As of January 1995, 29 commercial power units were operating in Russia in 9 nuclear power plants with a total installed capacity of 21242 MW. This figure includes 13 units with VVER (6 VVER-400 units and 7 VVER-1000 units), 15 units with uranium-graphite channel-type reactors (among them 11 RBMK units), and 1 unit with fast reactor (BN-600).

The main characteristics of NPPs are presented in Table 1. The NPPs generated 97.83×10^6 MW-h (119.185×10^6 MW-h in 1993).

The mean load factor was 52.53 % (67.22 % in 1993). This reduction of the load factor is explained mainly by grid limitation.

During 1994 there were 127 events at the 29 units in operation. As to the international estimate of the events, only 13 of the above 127 incidents can be classified by the INES: 1 incident - level 2, 12 incidents - level 1.

From 37 unplanned outages 11 involved the triggering of emergency protection systems. As a whole, in 1994 safety indices of NPPs in operation were higher than in 1993.

2. THE BN-600 NUCLEAR POWER PLANT OPERATIONAL EXPERIENCE

Operating histogram of the BN-600 reactor for 1994 is shown in Fig.1. As a whole, operation was successful. During this year there have been two shut-downs of the reactor for planned maintenance work and core refuelling. During this period the load factor was 78.2 % (80.3 % in 1993). This reduction of the load factor compared to one in 1993 is caused by increasing of heat output for consumers (0.49 %), longer planned repair works and refuelling period (0.66 %) and electrical grid limitation (0.53 %).

Table 1. Characteristics of NPPs in Operation. 1994

NPP Name	Unite N	Capacity MW(e) Gross	Electricity Generation GW-h	Load Factor %	Availability Factor %
Kola		1760	7304,7	47,38	51,28
	1	440	2139,3	55,50	57,10
	2	440	437,8	11,36	17,16
	3	440	2642,3	68,55	71,40
	4	440	2085,3	54,10	59,44
Novovoronezh		1834	7514,2	46,77	49,63
	3	417	2913,6	79,76	81,19
	4	417	2110,9	57,79	58,58
	5	1000	2489,7	28,42	32,74
Balakovo		4000	14662,6	41,85	70,21
	1	1000	1881,3	21,48	77,57
	2	1000	5148,4	58,77	83,53
	3	1000	3601,3	41,11	70,69
	4	1000	4031,6	46,02	49,06
Kalinin		2000	9437,7	53,87	54,67
	1	1000	4701,7	53,67	54,18
	2	1000	4736,0	54,06	55,17
VVER-440 units		2594	12329,2	54,25	57,26
VVER-1000 units		7000	26590,0	43,36	60,42
Kursk		4000	17764,9	50,70	51,75
	1	1000	1726,1	19,7	19,7
	2	1000	4665,0	53,25	53,44
	3	1000	5564,9	63,53	66,68
	4	1000	5809,0	66,31	67,18
Leningrad		4000	20323,7	58,00	64,12
	1	1000	6084,2	69,45	77,77
	2	1000	202,3	2,31	2,31
	3	1000	7266,7	82,97	92,91
	4	1000	6770,5	77,29	83,5
Smolensk		3000	16482,0	62,72	78,28
	1	1000	4655,1	53,14	71,33
	2	1000	5744,0	65,57	80,39
	3	1000	6082,9	69,44	83,12
RBMK units		1100	54570,2	56,63	63,48
Beloyrsky	3	600	4109,6	78,19	79,73
Bilibino		48	218,87	52,06	71,17
	1	12	60,11	57,19	86,32
	2	12	58,28	55,44	77,73
	3	12	53,99	51,37	72,44
	4	12	46,48	44,22	60,18
All units		21242	97818,37	52,57	62,19

In Table 2 main BN-600 reactor characteristics for the last five years and from the start of operation are presented.

In 1994 there were no water-into-sodium leaks in steam generators.

Table 2. Main BN-600 Reactor Characteristics for the Last 5 Years and from the Start of Operation.

Characteristics	Units	1990	1991	1992	1993	1994	From the start of operation up to Jan.1,1995
Electric power output	10^6 kWh	3464	3670	4402	3915	3810	53119
Load factor	%	65.91	69.83	84	80.3	78.2	69.2
The number of unit shut-downs		3	2	2	2	2	73
The number of loop outages		7	4	2	0	0	57

Table 3. Classification of Sodium Leak Development Causes.

Failure causes	Types of equipment			Total No/%	Over circuit	
	equip.	valves	pipes		I circuit	II circuit
Manufacturing defects	2	1	3	6/20.7	2	4
Design faults	-	-	5	5/17.3	3	2
Repair faults	-	8	2	10/34.5	-	10
Operational causes	1	1	-	2/6.8	1	1
Ultimate lifetime	-	6	-	6/20.7	-	6
N0/%	3/10	16/55	10/4.5	29/100	6/20.7	23/79.3

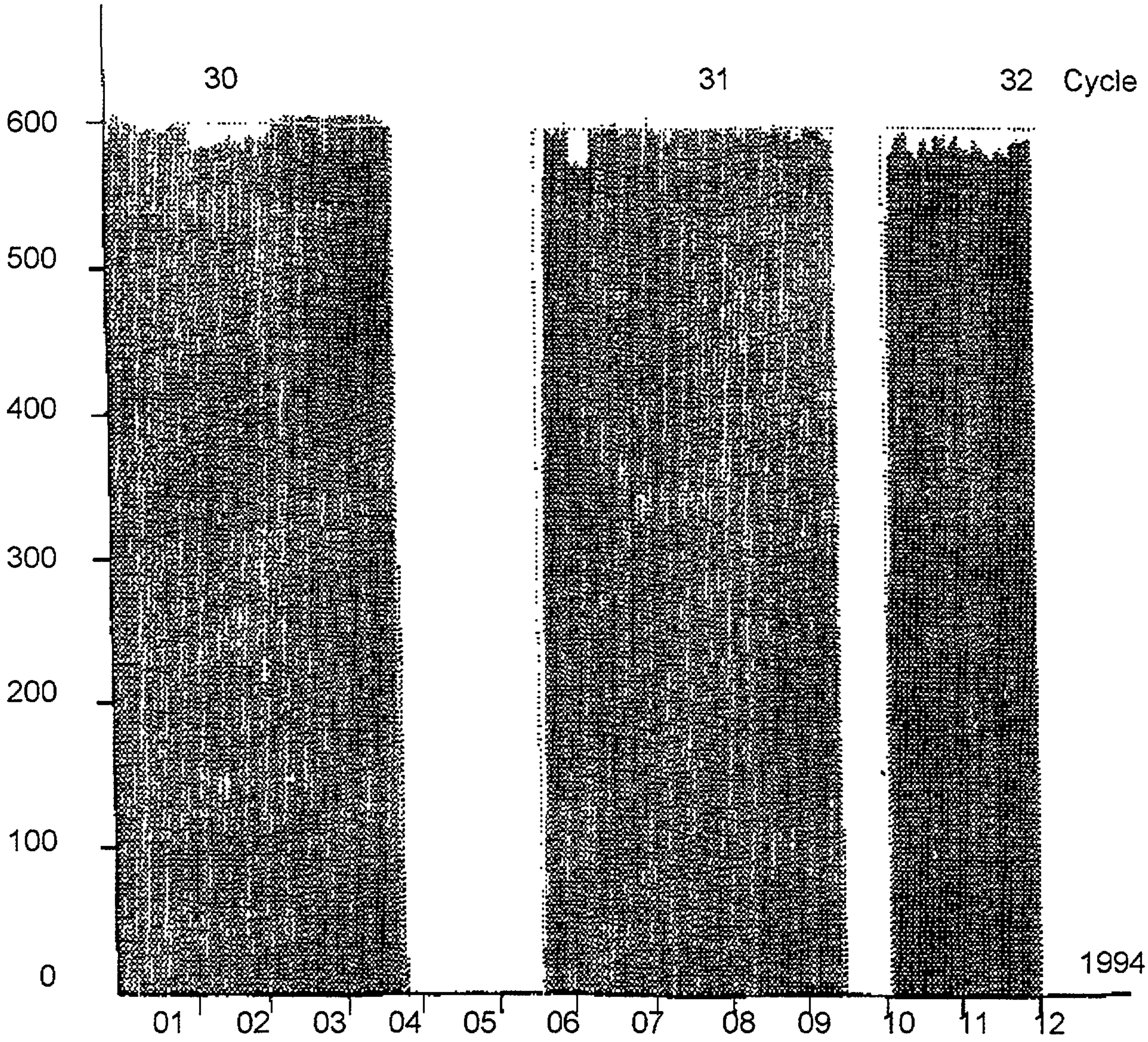


Fig. 1 BN-600 Operating Histogram.

Sodium leak on 06.05.94 in the intermediate heat exchanger drain line

During the planned outage while replacing the valve 1 (Fig. 2) jammed in the closed position the non-radioactive secondary sodium leak and burning occurred on the 48 mm diameter drain line (2) of the intermediate heat exchanger. The leak and burning continued for 9.5 hours and terminated when about 1.3 m^3 of sodium had spilled. The weight of the burned sodium was of the order of several tens of kilograms. The sodium remainder was retained in the metal catch pans (4) which were placed one by one underneath the leaking point and covered with the extinguishing powder. The room where the leak occurred had metal liner (5) and was equipped with the sodium drainage system of 100 mcapacity (leaking sodium to be dumped via the pipeline (6) to the drain tanks (7) located in the room) which was not used due to the small size of the leak.

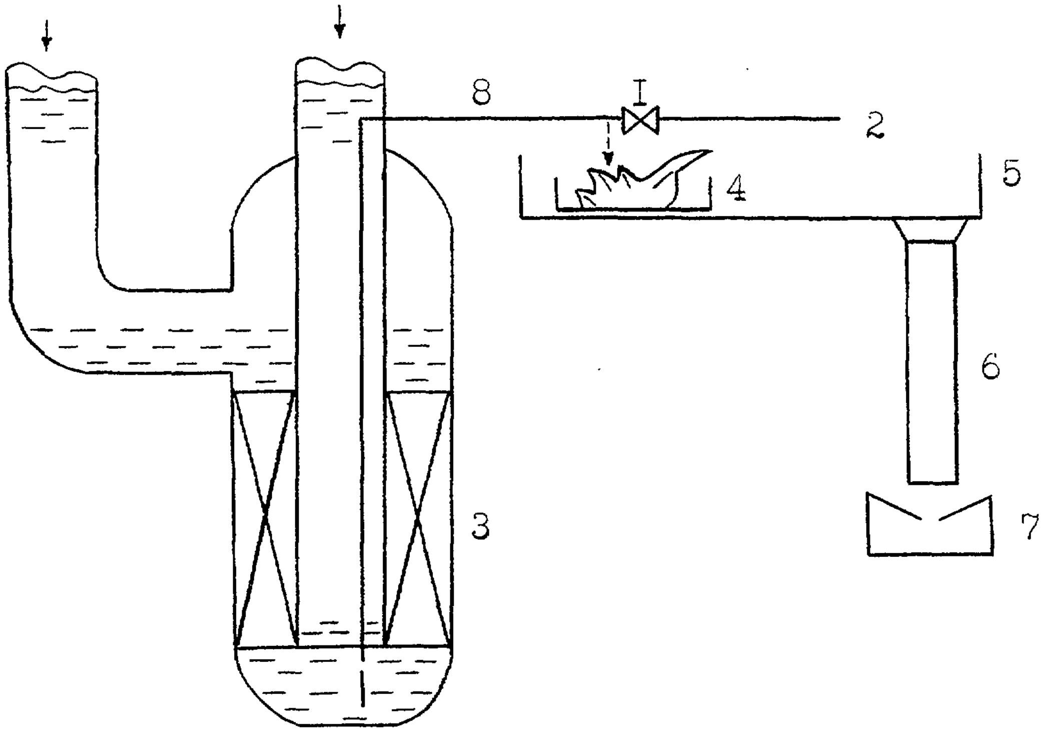


Fig. 2. Intermediate Heat Exchanger Drainage System.

The direct cause for the leak was a gas lock formed in the upper leg (8) of the drain line. The lock had low temperature thereby simulating the frozen sodium.

The reason of this incident was a fault during repair work. This event was estimated as level 1 on INES scale.

In Table 3 sodium leak statistics for a period of BN-600 operation and their classification are presented.

Operating experience with the core

As known, after the 26-th run the BN-600 reactor core has been changed over to the third charge with a maximum fuel burn-up of 10 % h.a., the maximum dose being 75 dpa. The fuel subassembly duct material is ЭП-450 type ferritic-martensitic steel (13Cr-2Mo-Nb-P-B), the fuel pin cladding material is the cold worked austenitic steel (16Cr-15Ni-2Mo-2Mn-Ti-P-B).

To confirm the reference fuel subassembly performance up to higher burn-up values by the end of 32 cycle irradiation of 63 fuel subassemblies was performed to a maximum burn-up more than 10 % h.a., in 20 subassemblies the dose reached more then 80 dpa; in 1 subassembly maximum dose - 80 dpa, maximum burn-up - 11.8 % h.a.; in subassembly the dose - 92.5 dpa, maximum burn-up - 11.2 % h.a. Among these 63 subassemblies only 1 fuel pin lost a tightness.

In the BN-600 core the tests of experimental fuel subassemblies with mixed oxide fuel fabricated using pellet and vibratory compacted fuel technologies are carried out. Currently, 4 subassemblies with pellet mixed oxide fuel are irradiated. By the end of 32 cycle the burn-up of these subassemblies has reached 10 % h.a.

Safety Analysis

In 1994 there have been finished activities related to obtaining the temporal license on the BN-600 operation. They are including as follows.

- Publication of "The Safety Assurance Report".

- Analysis of satisfaction with new safety regulatory documents, drawing up a list of deviations of NPP characteristics from requirements of these documents, drawing up a list of and carry out activities on obviating these deviations or on validating the compensating measures.

-Development of the Instructions on Beyond Design Basis Accidents Management.

These Instructions include the following accidents.

1. Loss of grid electric power supply with an attendant safety system failure.
2. Total loss of grid and emergency (diesel generators) power supply.
3. Guillotine-type rupture of primary circuit auxiliary system piping having no protective jacket.
4. Guillotine-type rupture of the main sodium pipe of the secondary circuit.
5. The formation of the hydrogen-air mixture in the steam generator cell.
6. A failure of the grid and emergency power supply with attendant safety system failure (ULOF).
7. Loss of tightness of the main and safety vessels of the reactor and fire in the vault.
8. Fire with a damage of monitoring and power supply systems.
9. Hydrogen - and/or - carbon containing composition ingress into sodium of a primary circuit.
10. Damage of constructions and equipment as a result of shock wave impact of 0.03 MPa, aircraft impingement upon the unit, earthquake of 7 points on MSK-64 scale.

11.Expulsion of the fuel subassembly from the core into upper plenum by the coolant flow.

3. R&D PROGRAM

3.1. PHYSICAL RESEARCH AT CRITICAL FACILITIES

3.1.1. THE BFS CRITICAL FACILITIES

The BFS-2

In the first half of 1994 there were measured characteristics of the fast reactor core in the center of which a representative insert of 400 l volume was set up containing inert diluent (aluminium oxide) - based fuel (the BFS-58 - assembly). In the insert fuel U-238 is absent and average fuel density is noticeably lower than that in the traditional cores. These circumstances result in substantial variation of neutron characteristics for the core composition.

In the center of the insert there were measured spectral characteristics - average fission cross-section ratios for 14 isotopes including minor actinides by means of several methods, and the ratios of capture cross-sections in aurum-197, neptunium-237 and uranium-238 to uranium-235 (or plutonium-239) fission cross-section.

There were also measured reactivity coefficients of main reactor materials samples, the sodium void effect of reactivity, the central control rod mock-up efficiency, Doppler effect at oscillating of neptunium-237 and plutonium-240 samples. The results of the experiments are still being analyzed now. For the central part of the core a negative value of the sodium void effect of reactivity was observed making up ~0.04 β_{eff} per 1 kg of sodium.

In the second half of 1994 a central insert of 7 BN-800 type reactor fuel subassemblies with mixed power-grade plutonium fuel was assembled at the BN-800 reactor mock-up with the reactivity compensation rod system being withdrawn.

The main aim of investigations on this insert was to obtain the SVER value for the core of real design and mixed power-grade plutonium fuel. Experiments conducted before were carried out with weapon-grade plutonium in a tube-and-pellets structure of the BFS.

At preliminary studies on an insert of fuel subassemblies not filled with sodium there were selected two subassemblies having close efficiency at placing them into the center of the insert. One of these subassemblies was not filled with sodium and was a reference one at determining the SVER in the insert of fuel subassemblies with sodium. The sodium void effect of reactivity proved to be equal to 0.013 β_{eff} that was satisfactorily confirmed by calculation.

The BFS-1

Investigations of the effect of neptunium upon the main neutron and physical characteristics of the core of a typical fast-reactor composition were carried out in cooperation with France.

In early 1994 about 250 pellets of neptunium dioxide were manufactured in each of which 40 g of neptunium was enclosed within a thin-walled stainless steel can of 5 cm in diameter and 1 cm thick.

Such amount allowed to replace some part of depleted uranium dioxide by neptunium dioxide in the central zone of ~30 l volume, neptunium content being 13.5 % heavy nuclei. Deliberately increased neptunium content allowed to sharpen its effect upon the parameters under investigation.

The initial core (BFS-67-1) represented a central zone with mixed fuel simulation with a weapon-grade plutonium content of ~19 %. The plutonium core volume was 300 l. The latter was surrounded by the driver uranium zone of 740 l volume with a specially selected composition, so that the effect of uranium fuel upon the central characteristics of the test zone was insignificant.

In the center of the original core and then in the core with neptunium there were measured by several methods, the ratios of average fission cross-sections for 16 isotopes, including minor actinides, and of capture cross-sections in aurum, neptunium, uranium-238, the central reactivity coefficients with the use of samples, the sodium void effect of reactivity, the efficiency of a mock-up of the central control rod with enriched and natural boron carbide, ad well as the fission reaction rate distributions with height.

A group of experimentators from the Nuclear Center at Cadarache, France, took part in the measurements of average fission cross-section ratios.

At Np insertion the core criticality varied less than by 0.001 $\Delta K/K$.

The results obtained indicate some marked relative increase of fission contribution from neptunium-237, plutonium-240, 242, americium-

241, 243, curium-244 - within 10-13 %, and from uranium-238 - more than 15 %. The positive component of the sodium void effect of reactivity in the center increased 3 times and the boron rod mock-up efficiency decreased by 5-15 %.

3.1.2. THE COBRA CRITICAL FACILITY

There have been carried out investigations concerning uranium-thorium cycle at four critical assemblies of the COBRA facility. The central inserts of these assemblies contained U-235, thorium and hydrogen. Hydrogen/U-235 ratio of nuclear density varied from 0.0 to 70.0. These experiments allow to estimate an influence of thorium on neutron spectrum and an accuracy of nuclear data in a wide energy range (from Mev to tens of Kev). In experiment there have been obtained Keff values of various uranium-thorium compositions, cross-section ratios of some nuclear reactions including thorium capture-to-fission ratio, fission cross-section of some TRU. On completion of the experiments the comparison of the results obtained with the calculated ones was made. The calculations were carried out using the following codes: KRAB, FFCP, HEEPC. Calculation results agree with the measured ones within 5-10 % for relatively hard neutron spectrum.

3.2. WEAPON PLUTONIUM UTILIZATION IN FAST REACTORS

In 1994 there have been analyzed two types of the BN-800 core:

- the core with higher fuel burn-up in the subassemblies a duct size of which is 96*2 mm;

- the core with Pu in an inert matrix without U-238.

In the first type of the core maximum burn-up increased from 10 % h.a. to 15 % h.a. herewith a subassembly duct size was changed from a design one of 94.5*5 mm to 96*2 mm. Fuel pin size was also changed from 6.9 mm to 6.6 mm. To guarantee higher burn-up (15 % h.a.) a fuel cladding thickness was increased from 0.4 mm to 0.55 mm.

One of the main requirements for advanced core development is ensuring of a zero sodium void effect of reactivity. For compensation of an additional positive component of reactivity arising due to higher burn-up a transition to axial heterogeneous core concept was made. Herewith the inner blanket of depleted uranium dioxide is placed in the central part of the core. A size of this blanket is 2/3 of the core radius.

This core concept uses fuel pins of equal enrichment. Inner axial blanket provides good flattening of an energy field distribution.

As a result of calculation analysis the following characteristics of S/A were obtained:

- in the upper part of S/A there is placed natural boron oxide bundle 150 mm high;
- a lower 127 - pin bundle consists of the core 880 mm high including an inner blanket of depleted uranium oxide 80 mm high, a lower blanket 350 mm high and finally a lower expansion helium chamber 670 mm high.

Both bundles have an independent junctions and are divided by sodium layer 300 mm high.

An analysis has given the following core characteristics

- gross thermal power (MWth) - 2100
- fuel pin linear power (W/cm) - 500
- maximum fuel cladding temperature (°C) - 700 (685 for ЭП-450 steel).

211 subassemblies with fertile layer are charged in the inner part of the core and 354 subassemblies without fertile layer are charged in the outer part of the core.

The required reactivity margin for burn-up is provided by plutonium fuel enrichment of 21.3 %.

Another advanced core under development is based on plutonium fuel without U-238 and oxygen (inert matrix).

3.3. STRATEGY OF DESIGN IMPROVEMENTS AND LOW RADIOACTIVE MATERIALS RECYCLING TO FACILITATE DECOMMISSIONING

An essential distinction of nuclear power from the power installations of other type is that at the end of a NPP's useful life a great amount of radioactive materials (even without consideration of nuclear fuel) is accumulated here. Radwaste management including final disposal creates many difficult problems. A large scale of these problems makes to revise the design practice priorities and to introduce for use some criteria reflecting radiation and ecology effects of used radioactive materials.

Construction of such criteria may be based on different aspects of radiation effects on a man, the environment and on the cost of the necessary safety measures.

Specific aim of investigation performed was to improve the existing design of BN-800 (variant 1) to facilitate decommissioning. Four optimization criteria were involved: the total activity of used reactor materials (without fuel subassemblies), the total radwaste quantity, the cost of safe waste management, re-use and final disposal.

Different ways of these criteria minimization were used: the search of low-level activated materials, new design approaches, cladding failures reduction, deep decontamination, recycling and re-use of activated materials under specific clearance levels. Two steps of optimization have been done.

On the first step (variant 2) design improvements were applied such as more wide use of boron carbide and its proper displacement, exclusion of neutronguides, more strict requirements to the level of cladding failures etc.

The second step was based upon perspective design improvements: wide application of titanium, titanium hydrides, boron carbide, chromium-manganese steel with restricted to 0.01 % cobalt-59 content; exploitation of reactor without cladding failures etc. The design experience and published results of French, British, German, Russian and other countries specialists on LMFR were of vital importance in the work. Some of the considered design improvements are still rather problematic and a lot of technical and technological elaboration's must be done to realize them.

The Fig. 3 represents the final results of the decommissioning costs radioactive component optimization. This component included: costs of

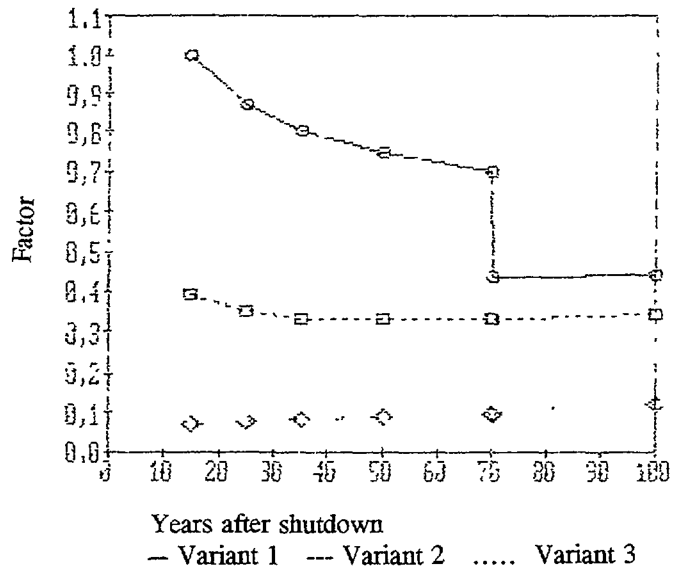


Fig. 3. Relative Dependence of the Decommissioning Costs Radioactive Component.

radiation exposure risk, of final radwaste disposal, of used materials management and refurbishment, of radioactive control and safety measures, profit from returned valuable materials. It was reduced considerably.

There is no more need in the variant 3 for "a long cooling-down period" because radiology risk (and cost for it) is very small and most part of used materials are under the level of unrestricted release immediately after shut-down.

Moreover, almost all used materials in variant 3 admit re-use in the similar reactor for several times. Hence, we can organize in nuclear industry the Closed Cycle of used materials and due to that not only to ensure reliable market of low-level activated materials but to realize the "Non-proliferation Principle" for them.

3.4. MOLTEN FUEL - COOLANT INTERACTION INVESTIGATIONS

In 1994 two experimental runs have been carried out on PLUTON installation (Fig. 4). Simulation experiments nave been effectuated using the following pairs of interaction media: alumina/sodium in first run and thermite mixture (Zr + Fe_2O_3)/sodium in second run.

Alumina was used in experiments with single fuel element. These experiments were performed under the following conditions:

- mass of the melt - 6-12 g;
- temperature of the melt - up to 2700 °C;
- sodium temperature - 420 °C;
- sodium volume - 5 l.

There was not observed violent interaction (vapour explosion). Interaction resulted in a fragmentation of the melt with dominant particle size of 100 mkm.

On second experimental section 7 fuel element bundle filled with thermite mixture ($Zr+Fe_2O_3$) was used.

The conditions of the experiments were as follows:

- mass of the melt - 0.165 kg, 0.488 kg;
- temperature of the melt - 2500 °C;
- sodium volume - 4 l.

MFCI was accompanied with intensive acoustic noises and pressure pulsations without time delay. Mean size of the solidified particles measured was 240 mkm.

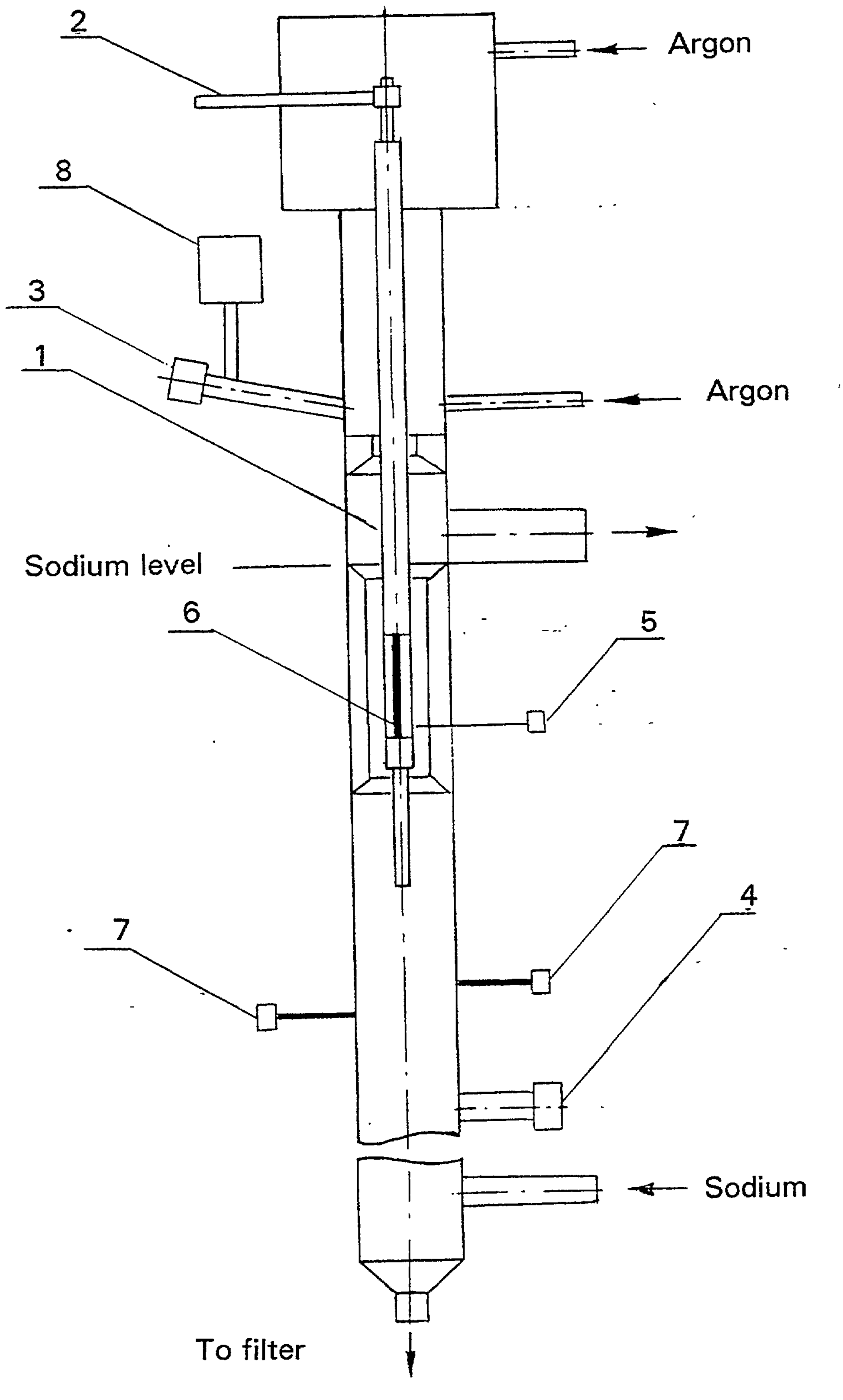


Fig. 4. Experimental Section (PLUTON).

1 - fuel element imitator; 2 - current supply; 3 - pressure transducer;
4 - accelerometer; 5 - acoustic transducer; 6 - fuel element body;
7 - thermocouples; 8 - level indicator.

An analysis of a set of the data on the dynamical characteristics of interactions has shown an essentially milder interaction energetics in the case of sodium coolant as compared with water.

It is planned to continue investigations of MFCI in conditions near to real subassembly fuel bundle.

3.5. EXPERIMENTAL INVESTIGATION OF SODIUM FIRE

In 1994 some experiments simulating hypothetical accidents with instantaneous rupture of pipeline has been carried out.

A wealth of statistical data have been obtained for the sizes of leakages possible and its development for more than 40-year period when sodium has been actively used as a coolant.

The results stated give grounds for some conclusions concerning this problem and approaches to designing sodium systems for fast reactors. In brief they are as follows:

- not a single case of guillotine-type rupture has ever taken place in reality. Therefore this accident was considered to be incredible;

- the leakages (more than 50 events total at home-made reactors) were mostly short duration and drop type, or of very small flow rate. Only few cases of considerable duration (up to 10-15 hours) have taken place, with sodium leakage flow rates up to 100 kg/h and sodium outflow amount up to 1500 kg;

- no increase of the defect size has been detected during the accidents. On the contrary, sometimes the "self-healing" phenomenon was observed, and leakage stopped;

- leaks with extensive sodium spraying can take place in microfractures and at high pressures and they do not cause any dangerous consequences in the aspect of NPP safety.

However, the well-known facts such as increasingly strict safety requirements, the accident occurred at the Almeria facility, as well as rearrangements in the secondary circuit cells of the Super-Phenix reactor gave an impulse to additional consideration of potentially more intensive sodium fire in the incredible accidents with instantaneous ruptures of pipelines. In this connection, modeling studies on this problem have been started at the IPPE. The first experiments with sodium have been carried out. They have been specially made under the most adverse conditions: instantaneous rupture, vertical upward jet, obstacle at the jet route, outflow stream velocity up to 25 m/sec.

After sodium was heated (up to 400 or 510 °C) and membrane was broken at the pressures of 4.5 to 6.5 atm, sodium was injected into the steel vessel of 2.0 m^3 volume. Steel screen was installed inside the vessel; it was equipped with thermocouples and pressure transducers placed in different parts of the model. The time of sodium outcome from the measuring device was determined by electromagnetic flowmeter. The amount of sodium injected varied from 400 to 2000 g. The time of injection was 0.07 to 0.7 sec.

In the course of experiments the rate of pressure increase in the chamber was registered in the range of 4 to 24 atm/sec, and maximum gas temperature in the area of jet collision with the screen amounted to 1250 °C. Maximum pressure in the vessel reached 4.1 atm when 2000 g of sodium was injected and 1.6 atm when 400 g of sodium was injected.

It is planned further to extend the parameters range in order to cover more realistic conditions area. Preliminary analyses shown that 5-10 % of sodium can be sprayed in the case of large rupture. After completion of the experiments planned detailed analysis is scheduled.

3.6. STUDIES ON HYDRODYNAMICS AND HEAT EXCHANGE

Safety problems present the primary direction in research on heat exchange in fast reactors.

Work on an investigation of boiling in the core under conditions of decay heat removal with natural convection is continued. Experiments were conducted on an 7 pin bundle cooled by eutectic sodium-potassium alloy.

At a pressure of 0.06 MPa coolant boiling started at a heat flux density of 117000 W/m^2 (coolant flowrate through bundle subassembly being 0.75 m^3/hr). At a heat flux density less than 133000 W/m^2 there was observed a steady process of heat removal due to coolant boiling. At an increase of the heat flux density up to 150000 W/m^2 the process of boiling became pulsating, flowrate periodically decreased almost to zero, then sharply increased, splashes of wall temperature up to 90 °C were observed.

It is explained by periodical vapouring of the heated section: after filling the heated section with vapour the vapour plug floated up and the heated section was again filled with fluid.

There was carried out a work on mounting a new sodium loop for a study of sodium boiling at a 19 pins bundle (Fig. 5). Trial circulation of coolant in the circuit was carried out.

Experimental studies of the effect of blocking some part (55 %) of the flow section upon the velocity and temperature fields in a fuel subassembly were continued. The measurements of longitudinal and transverse velocity components have been made by the electromagnetic method with using a local-action sensor. The results obtained have been compared with the corresponding data for a model with smooth pins.

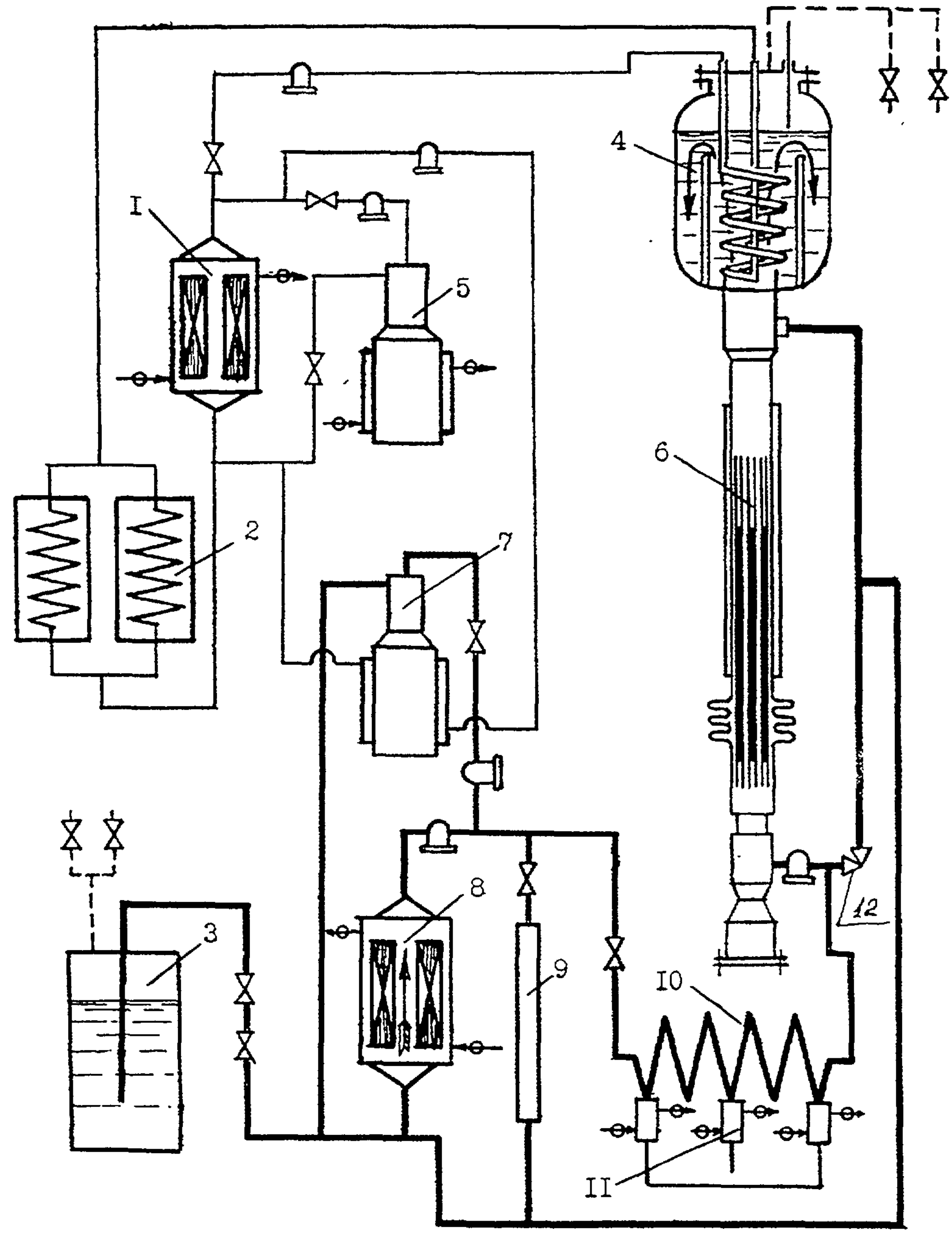


Fig. 5. Technological Diagram of the Experimental Loop.
1-EMP of the cooling circuit; 2-air cooler; 3-dump tank; 4-cooler; 5,7-cold traps; 6-experimental mockup; 8-EMP of the sodium circuit; 9-calibration section; 10-preheater; 11-electric supply.
——— sodium circuit pipe lines; ——— cooling circuit pipe lines;
– – – – – gas-vacuum pipe lines; —⊖➤ water supply.

With helical wire spacers there is practically observed no reverse flow, the process of velocity recovery begins earlier than in case of a blocked subassembly with smooth pins (Fig. 6).

The transverse component profile for all the pins studied is characterized by non-uniformity the value of which is increased with Reynolds number.

Thermal experiments were carried out on a bundle with 37 pin arranged in a triangular lattice with a pitch of 1.17 and spaced by helical wire wrap.

Temperature field is formed under the influence of flow hydrodynamics: there is observed a sharp rise of the central pin wall temperature (Fig. 7) prior to blockage, a peak immediately after blockage, a temperature drop at the section of velocity recovery. A maximum level of azimuthal irregularities of fuel pin temperature exceeding the nominal one by 1.5-4 times is located in the behind-blockage area (Fig. 8).

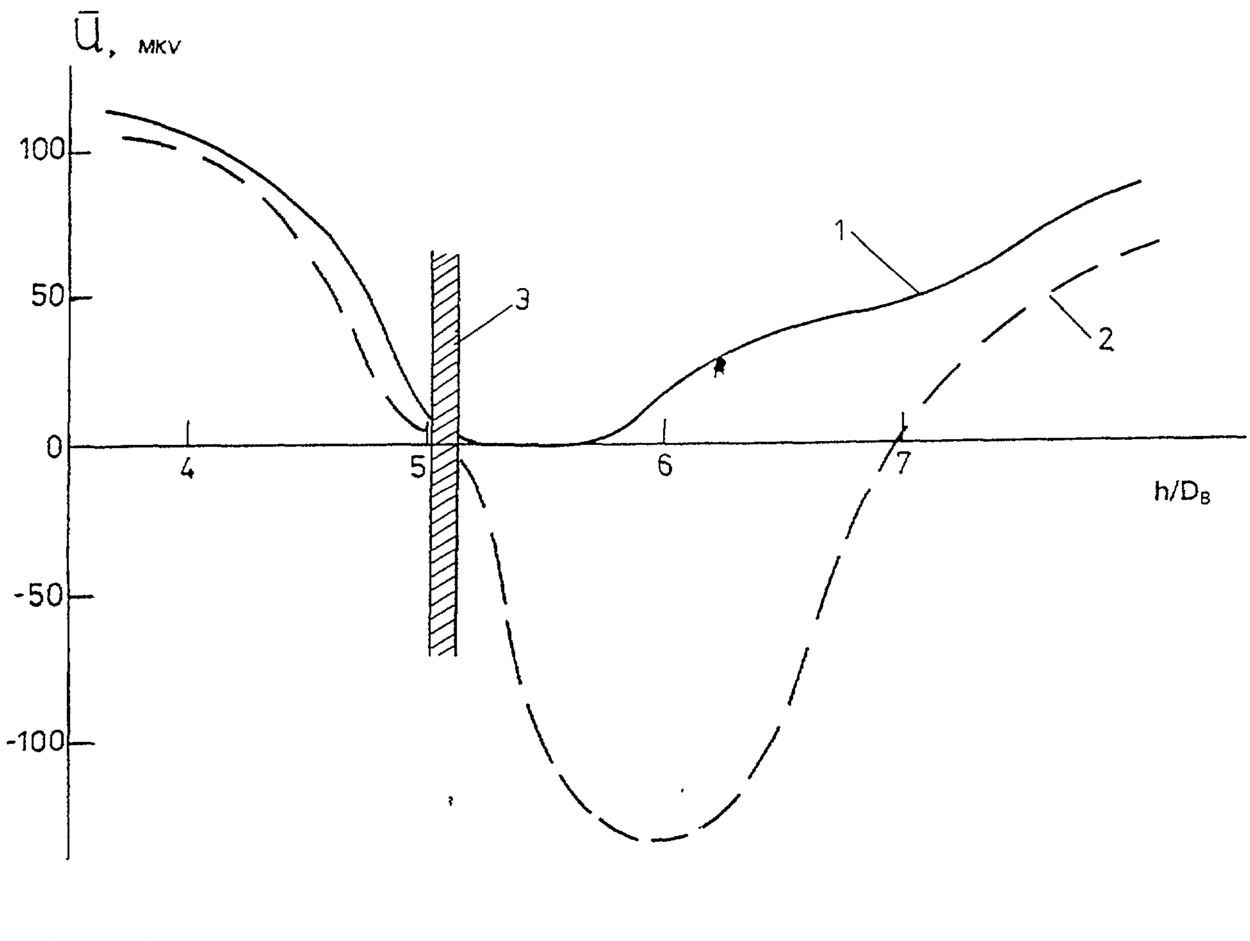


Fig. 6. Velocity Profile Component Distribution along the Central Channel Length:
1-the central channel, pins with helical wire wraps; 2-the central channel, an assembly with smooth pins; 3-blockage.

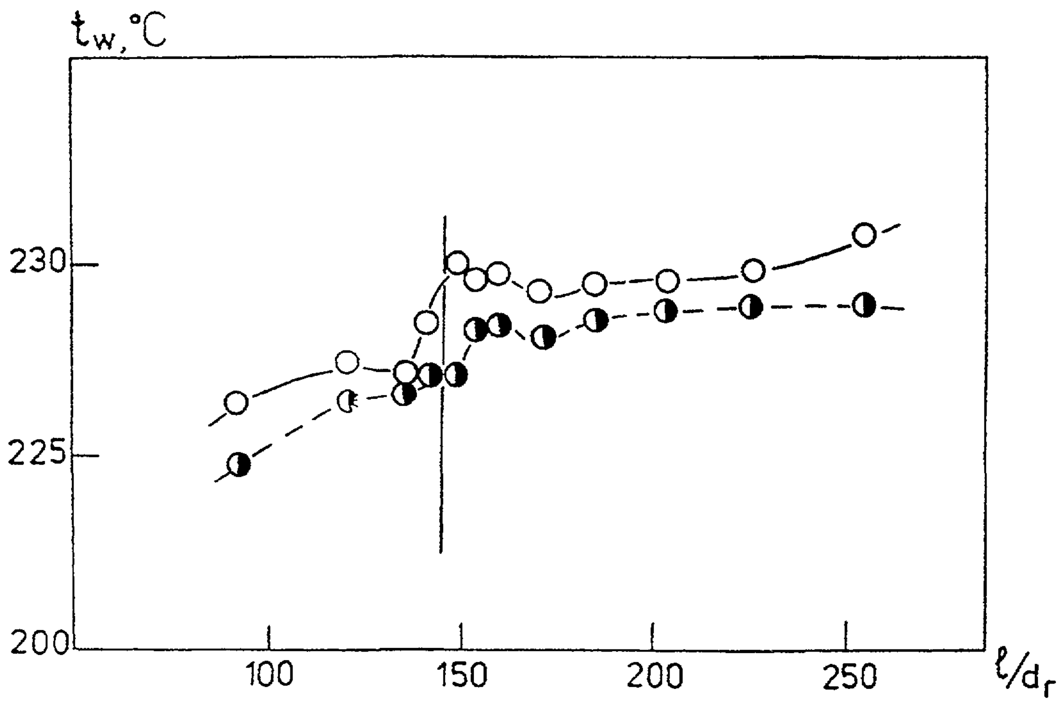


Fig. 7. Temperature Distributions with Height of the Central Dummy Pin at Pe=326.

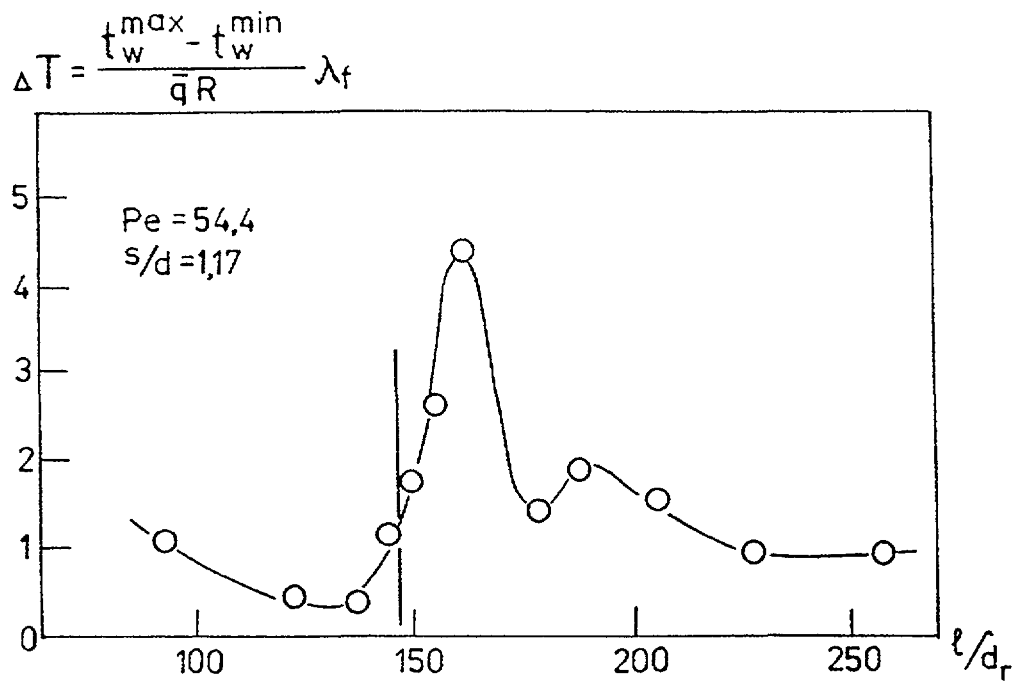


Fig. 8. Variation of Maximum Irregularity along the Central Pin Length.

3.7. FAST REACTOR ECONOMICS

The period of the end of 80-ies and the early 90-ies is characterized by the reduction of oil prices and some rise in the NPP costs. In addition, the problem of uranium deficiency has been shifted to the beginning of the next century. In this connection, fast reactors development in the nuclear power system will depend significantly on their economical competitiveness (unless environmental and non-proliferation problems will become first and foremost to be solved using FRs).

First semi-commercial FRs were behind the LWRs since their specific capital costs and electricity costs were 1.5 to 2.5 times higher. This was because of difference between the NPP types compared with respect to the extent of their industrial development.

Forecast evaluations show the world trend of bringing fast reactor economical characteristics nearer to those of thermal reactors when shifting to the reactor series construction and optimization of FR technical approaches.

It is worth-while to note that there are some features of fast reactor economics in Russia. On one hand, in Russia the BN-800 design development has been completed, its construction has been already started, and studies have been made on the improved (with respect to technical and economic parameters) BN-600M reactor. On the other hand, intensive works are under way on designing of thermal reactors of improved safety (NP-500 and VPBER-600 designs).

Increasing safety requirements favours fast and thermal reactors economical parameters rapprochement, since the cost of required safety level assurance has been already taken into account in fast reactor designs (for example, the BN-800 design). As it can be seen from the Table 4, VPBER-600 design material consumption has increased as compared to the series VVER-1000 reactor and approached the BN-800 design value. Further improvement of fast reactor technological system and equipment (BN-600M design) results in equalization of characteristics of pre-production fast reactor and series VVER-1000 reactor.

Calculation evaluations have shown that construction of three BN-800 units at the South-Ural NPP site and of three NP-500 units at the Kola NPP site would practically equalize their capital costs (series factor).

Table 4. Comparative characteristics of FR and VVER reactor facilities

Parameter	BN-600	BN-800	BN-600M	VPBER-600	VVER-1000
Thermal power (MW)	1470	2100	1520	1800	3000
Reactor facility mass (t)	7875	8435	6725	6085	8540
Specific material consumption of reactor facility (t/MWe)	13.0	10.5	10*(8.23)	9.7	8.44

* Integral steam generator option

The same result can be achieved by the Beloyarskaya NPP expansion by means of construction of one BN-800 unit or two NP-500 units, when factors of sites similarity and their equal availability for the NPP construction are taken into account.

4. NEAR-AND-LONG TERM PROSPECTS

As regards the near-term perspective (10 to 15 years) it can be stated that the advantages of fast rectors will not be completely claimed. First of all this concerns FR capability for nuclear breeding.

Fast reactor inculcation upon the nuclear power for only purpose of electricity production is also low probable in the near-term perspective, since there is sufficiently developed technology for light water reactors, and, besides, fast reactors in general still have worse economical characteristics.

Fast reactor construction in the nearest decades will be justified if they will be used reasonably as NPPs the safety level of which meets up-to-date requirements, and also for solving environmental problems, concerning spent nuclear fuel and released weapons grade plutonium.

At the same time, once should proceed from the fact that the generation using "nuclear electricity" should solve the problem of the produced wastes long-term management by their own forces, and should not shift these troubles to the shoulders of their descendants.

Fast reactor inculcation upon the nuclear power is also necessary if any country is guided by utilization of closed fuel cycle, when reprocessed uranium and plutonium serve as the efficient fuel for the NPPs with different reactor types.

Fast reactor real state-of-art and chosen strategy for nuclear fuel cycle will determine FR development in any country for the nearest future.

If the strategy of deep geological isolation of the irradiated fuel without its reprocessing is chosen then it can result in certain delay in fast reactor construction.

There are certain features of fast reactor development in Russia:

- reprocessing plants for uranium and power plutonium from the spent fuel are under operation in the country (fuel from VVER-440, transportation facilities and fast reactors), and weapons grade plutonium surplus is released. It should be also taken into consideration that partial replacement of uranium with plutonium as NPP fuel is highly profitable for Russia, since selling of part of uranium in the external market is possible;
- fast reactors in Russia are now considerably more prepared from the technological standpoint than light water reactors for plutonium utilization (BR-10 and BOR-60 reactors operation with plutonium loading and MOX fuel subassemblies tests in the BN-600 reactor);
- economical characteristics of FR NPP designs under development now (BN-800) are practically nearly the same as those of new LWR designs (NP-500, VPBER-600);
- finally, there is considerable positive experience of various power fast reactor development, construction and operation (96 reactor-years) gained in Russia.

Taking into account the above mentioned factors and in accordance with the worked out concept of nuclear power development until 2010 for the next stage of fast reactor development in Russia construction until 2010 of 3 or 4 BN-800 units using uranium-plutonium fuel and plant for fuel subassemblies fabrication (PO "MAYAK") is planned. At the same time solution of electricity production problem as well as of the whole scope of environmental problems is provided including civil and weapons grade plutonium utilization.

It is difficult to make long-term forecasts of development of nuclear power as a whole and in particular of fast reactors. In this connection it is impossible to determine for sure the time in the future, when fast reactors will be needed operating in the nuclear fuel breeding mode. This will depend on development rate of power engineering (including nuclear power) in different countries.

Nevertheless, taking Russia as an example let us try to determine approximate time, when FRs could widely inculcate upon the nuclear power in connection with the necessity of its supply with nuclear fuel.

Table 5 contains various strategies of Russian nuclear power development until 2010 and later. Practically all options of development until 2010 do not impose any problem with nuclear fuel. However, if the development of nuclear power is rather intensive until 2010 and this rate is maintained until 2030 (almost four times increase of NPP power level as compared to that of today), then the available fuel base will not be sufficient.

Apparently, realization of this scenario of the nuclear power development in Russia after 2015 will require more extensive inculcation of fast reactors using their fundamental feature: nuclear fuel breeding.

Table 5. Installed Power of NPPs in Russia before and after 2010 (GWe)

<table>
<tr><th>Options before 2010</th><th>1995</th><th>2000</th><th>2005</th><th>2010</th><th>2015</th><th>2020</th><th>2025</th><th>2030</th><th>Options after 2010</th></tr>
<tr><td>Minimum</td><td>23.3</td><td>23.3</td><td>21.6</td><td>25.5</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Medium</td><td>23.3</td><td>26.5</td><td>28-31</td><td>30-34</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td rowspan="3">Maximum</td><td rowspan="3">23.3</td><td rowspan="3">26.5</td><td rowspan="3">30.7</td><td rowspan="3">39.5</td><td>31.9</td><td>28.9</td><td>29.4</td><td>21.9</td><td>Curtailment of nuclear power as NPP are decommissioned</td></tr>
<tr><td>39.8</td><td>40</td><td>40</td><td>40</td><td>Maintaining power level as of 2010 during 40 years with subsequent cessation of NPP construction</td></tr>
<tr><td>49.9</td><td>60.3</td><td>70.5</td><td>80.5</td><td>Achievement of 80 Gwe level by 2030 and its maintaining until 2050 with subsequent cessation of NPP construction</td></tr>
</table>

Providing with fuel of different options of nuclear power development in Russia

1. All options of nuclear power development until 2010 are fully provided with the storehouse stocks of uranium.
2. Option of curtailment of nuclear power after 2010 is also fully provided with the storehouse stocks of uranium.
3. Option of nuclear power level of 40 GWe maintained after 2010 uses all uranium stocks together with the released weapons grade uranium.
4. Option of 80 GWe power level achievement by 2030 is not provided by the existing raw material base.

A REVIEW OF THE UK FAST REACTOR PROGRAMME

K.F. AINSWORTH
British Nuclear Fuels plc,

C. PICKER
AEA Technology

Risley, Warrington, Cheshire,
United Kingdom

Abstract

The general position with regard to nuclear power and fast reactors in UK is described, including the current status of new installations, such as the Sizewell B PWR. The project arrangements now in place for fast reactor studies in UK are outlined and a description and progress statement for the programme of studies associated with the closure of the Prototype Fast Reactor is included.

A Review of the UK Fast Reactor Programme

1. General UK Situation

The latest available statistics on electricity supply in the UK pertain to 1993[1] and show total electricity generated was 323 TWh. This was an increase of 0.5 per cent on the previous year which has been about the annual rate of growth over the period 1989 to 1993.

The mix of plant used to generate this electricity has been seen to change. Nuclear stations generated 17% more than in 1992 while CCGT plant generated almost seven times more electricity in 1993 than in 1992. There was also an increase of nearly 50 per cent in electricity generated from renewable sources other than hydro-electric stations. Generation from conventional coal fired stations fell by 13 per cent. The breakdown of UK electricity production in 1993 by source was approximately: Coal 54%, Nuclear 27%, Gas 10%, Oil 7% and Others 2%.

Current UK nuclear capacity totals 14,000 MWe (Gross) the largest proportion of which is provided by Advanced Gas Cooled Reactors (64%) followed by Magnox Reactors (27%) and Pressurised Water Reactors (9%). This capacity is currently operated by the two Utilities; Nuclear Electric (78%), Scottish Nuclear (19%) and the fuel cycle company BNFL (3%).

Whilst 1994 had a sad start, particularly for fast reactors with the closure of the Dounreay Prototype Fast Reactor (PFR) the period since has seen many successes for the UK nuclear industry. The Utilities efforts have been rewarded with significant improvements in the operation efficiency of existing reactors and the UK's first Pressurised Water Reactor at Sizewell B achieved criticality on 31 January 1995. Good progress has also been achieved with a number of fuel cycle facilities which will make a major contribution to nuclear fuel recycle, ie:

[1] Digest of United Kingdom Energy Statistics 1994, DTI. A publication of the Government Statistical Service.

- commissioning of the Sellafield Thermal Oxide Reprocessing Plant (THORP)
- commencement of commercial irradiation of fuel manufactured in the Sellafield 8t/a MOX Demonstration Facility (MDF)
- commencement of construction of a commercial scale, 120 t/a Sellafield MOX Plant (SMP)
- good progress towards operation of the Springfields New Oxide Fuel Complex (NOFC) which has capability for fuel manufacture with recycled Uranium.

The major event which is awaited in the UK is the outcome of the UK Government Nuclear Review which has been in progress since last year. All the UK nuclear organisations have participated in the review by making appropriate submissions. A government announcement is expected in the very near future which may lead to some changes in the industry. The implications of the ongoing reorganisation of the UK Atomic Energy Authority are referred to later in this paper.

2. Fast Reactor Project Arrangements in the UK

In November 1992 the UK Government announced that responsibility for UK funding for fast reactor R&D would be transferred totally to the nuclear industry. The Government believed that the technology was matured to the stage at which Government funding of further development was no longer appropriate. The UK industry which regards the fast reactor as a long-term strategic requirement for development as a partnership between Government and Industry responded that it could not accept the additional cost to continue R&D activities at the then current level alone, or to operate PFR.

The Industry, however, recognised that total withdrawal from fast reactor development would not be in line with its longer term strategic interests and has jointly agreed funding to maintain an active but lower level UK participation in the European Fast Reactor Project. UK participation in the EFR project has been maintained under this arrangement since late 1993.

The only remaining UK Government funded activity which contributes to the European co-operation is the PFR closure experiments on sodium reactions and materials examinations. This work now only has a small residual funding requirement and will be completed by early 1996.

The UK Government is continuing to fund the PFR decommissioning and the completion of PFR fuel reprocessing.

The funding provided by the industry in 1995 will maintain a UK contribution to fast reactor development in the following areas:

- participation of BNFL and NE in the activities of the European Fast Reactor Utilities Group (EFRUG) and in Fuel Cycle Strategic studies,

- NNC engineering support to the EFR Associates reactor design studies in the areas of core physics, safety, ISIR, containment, DHR and feedback of PFR experience,

- UK involvement in the EFR R&D studies, particularly on the CAPRA project in the areas of core physics, safety and fuel performance involving AEA Technology, BNFL and NNC,

- maintaining international links including secondees to MONJU, MASURCA and EFR Associates in Lyon,

- support for IWGFR activities (such as the recent review of the safety of BN350 in Kazakhstan.

3. UK Contribution to EFR Design

Within the EFR Associates design organisation the UK continue to contribute to the collaboration with the support of the UK EFRUG members (BNFL and NE). This contribution consists of a permanent member of the EFR'A central team located in the offices of Framatome (Lyon) supported by a small team of designers in the offices of NNC. This contributes to the retention of a core team of fast reactor expertise.

The strategic role of fast reactor in the fuel cycle and the broadening of collaboration (outside Europe) are important to sustaining the European capacity. Consequently the UK contribution is directed at relevant topics. These include the safety of FR, core safety (including Pu burner cores), decay heat removal and containment and the design for inspectability/repairability. The feedback of operating experience to the EFR design is also important to build the utility's confidence in FR and a particular emphasis is being put on ensuring the feedback of PFR experience.

A comprehensive report on the EFR design activities to the IWGFR will be provided by M. Le Rigoleur.

4. Status of Prototype Fast Reactor (PFR)

4.1 Comment on Position of UKAEA

As part of the preparation for the privatisation of part of the former United Kingdom Atomic Energy Authority, the organisation has been divided into two main parts: AEA Technology, which is to become a privatised company offering technical services on a commercial basis to other organisations, mainly in the nuclear sector but also in non-nuclear markets, and UKAEA, which will remain Government-owned and which will be responsible for discharging the decommissioning of nuclear facilities and for radwaste management arising out of the past activities of the UK Atomic Energy Authority.

The Bill to establish AEA Technology as a wholly-Government-owned private limited company is expected to be ratified by the UK Parliament in July or October 1995, prior to privatisation in 1996.

4.2 Decommissioning and Fuel Reprocessing

As was reported in 1994, operation of PFR ceased as scheduled at end March 1994, following a very successful final year of operation. At that time, the reactor was handed over to UKAEA-Government Division to begin decommissioning.
Reprocessing of the PFR fuel has continued. A total of 10 core assemblies, 19 radial breeders and 456 kg of U + Pu fuel fabrication residues were processed in the 1994/95 financial year. In total 4048 kg of U + Pu (327 kg Pu) was processed.

The decommissioning of PFR is to be divided into three active stages separated by periods of care and maintenance. The first stage comprises initial decommissioning and preparation for care and maintenance and will last until end March 2001. An initial sub-stage will be to defuel the reactor, drain down the secondary sodium circuits and the NaK-filled decay heat removal circuits, isolate non-essential services and remove redundant plant. During the defuelling, the 117 core components will be replaced by dummy components. This initial sub-stage will take until end March 1996.

The second sub-stage, lasting until end March 2001, comprises the operation of a purpose-built sodium disposal plant, the removal of primary circuit sodium, conditioning of the primary circuit for long term storage, draining and cleaning out the buffer store and dismantling non-essential plant and buildings. On the present schedule, the plant will then be placed in long term care and maintenance, including surveillance of the integrity of the secondary containment building, until 2070.

The Stage 2 decommissioning, scheduled to begin in 2070, comprises dismantling of the secondary sodium and steam circuits, constructing a new containment around the primary circuit and irradiated fuel cave and dismantling the original secondary containment building. This is scheduled to take 6 years. The Stage 3 decommissioning, comprising dismantling of the reactor internals and the concrete shielding, will follow a further period of care and maintenance and will take 20 years.

4.3 Closure Experiments

The closure of the reactor and its subsequent decommissioning provided the opportunity, which may not recur, to conduct various tests and to obtain information of potentially inestimable value. AEA Technology, with the support of its industry partners in UK and EFR partners in Europe, drew up a number of proposals for work to be undertaken around the time of reactor closure. The aim of these was to be of value to future fast reactor development and, in particular, to the future plans of our European partners and for the support of the continued operation of the French fast reactors, particularly Superphénix. The programme also had the support in principle of the then Department of

Energy, although the Department advised that it could only support financially a significantly smaller programme than had been proposed initially.

The announcement in November 1992, that the Department of Trade and Industry (successors of the Department of Energy) would cease funding of the EFR R&D programme after March 1993, caused some revision of the scaled-down closure programme proposals so as to include work in some areas for which the European partners were wholly dependent upon the UK, but it was only late in December 1992 that the amount of funding to be provided by DTI became known. This resulted in a further scaling-down of the programme to its present level.

The final programme was in three parts:

i) Steam generator leak detection studies in PFR
ii) Sodium-water reaction tests in the Super Noah rig
iii) Destructive examination of materials from PFR

The first two studies took place in the 1993/94 and 1994/95 financial years and were completed on 31 March 1995. The programme to examine materials began in the 1994/95 financial year and will be completed by end March 1996.

4.3.1 Steam Generator Leak Detection Studies

Acoustic and hydrogen detection methods were examined for their performance during long term background measurements, made in the final periods of operation, and during detecting deliberate gas and steam injections made after operation had ceased. A total of 85 argon, 53 water and 4 hydrogen injections were made and were used to assess the sensitivity and speed of response of detection systems. The aim of these long term stability and injection studies was to examine whether a practical leak detection system could be developed from the acoustic and hydrogen detection methods.

Two types of acoustic leak detection were employed. These were a passive system, based on listening for the acoustic signal produced by a leak and an active system based on attenuation of a transmitted pulse by the gas produced by a leak. This latter method was proposed by CEA in France and was carried out by members of their staff. It is not reported here.

The conclusion from the passive acoustic detection studies was that the equipment used was suitable for an operating leak detection system. However, a good understanding of the noise producing events during start-ups and shut-downs would be essential if spurious trips of the plant are to be prevented. Some difficulties might be expected in locating leaks at the heart of a tube bundle because of the transmission of acoustic signals through the structure, as well as directly to the detectors. An unexpected result was that the acoustic signal reaches a plateau with increasing leak rate so that detecting a water leak of 20 g/s or even 30 g/s would not be significantly easier than detecting a 5 g/s leak.

Three mass spectrometers were used for the hydrogen detection studies, one on the gas spaces, one for the sodium phase on the existing hydrogen detection loop on the outlet from Evaporator 3 and the third on a new loop a the sodium inlet to Evaporator 3. Unfortunately, the detection membrane on this new loop was damaged on installation and could not be used. It was therefore not possible to assess the stability/availability of a system based on measuring hydrogen concentrations at the inlet and outlet of a SGU. It was concluded that mass spectrometers are more stable than the installed katharometers at PFR. The evidence suggested that the dissociation of NaOH at 350°C is slow and can be discounted when assessing the initial response of in-sodium hydrogen detection systems.

The initial rate of hydrogen ingress was shown to be very high and to persist for several days following starting up from a long shutdown, during which the magnetite film integrity on the inside of ferritic SGU tubes had partially degraded. This compares with a very brief transient during water/steam admission in the absence of magnetite degradation. Future trip systems based on hydrogen detection must be able to take this into account.

Examination of the system pressure and sodium flow was also made, to examine their potential use as a leak detection system. The studies showed that these parameters were broadly consistent with those calculated using the QUARK pipe network modelling computer code, but could only be used to detect very large leaks.

4.3.2 Sodium-Water Reaction Tests in the Super Noah Rig

Following a multiple-tube failure in a PFR superheater in 1987, two models were proposed to place deterministic limits upper limits on the number of tube failures following an initial leak. These were a reaction flame heat transfer model, to show that a large number of tubes could not be brought to the point of failure at the same time, and a 'piston expulsion' model, where most of the sodium would be expelled from the SGU, blanketing the tubes with steam and stopping the sodium water reaction when a certain number of tubes had failed.

Tests were made in the Super Noah facility to validate these models and the two phase reacting sodium water mixture model in the QUARK computer code.

Tests on a PFR evaporator tube configuration in the Super Noah rig demonstrated piston expulsion of sodium will dominate after an incident has escalated to about 30 double ended guillotine failures. The cooling effect of the steam will then make it extremely unlikely that further tubes will fail. Local pockets of sodium entrained within the steam flow or in fairly stagnant regions of the SGU might, however, generate very high temperatures and cause other tube failures over a period up to 10 seconds. These failures will be limited in number and localised.

A further experiment investigated the heat flux through tubes in different regions of a sodium water reaction flame from an intermediate size leak in a PFR evaporator. It was concluded that wastage was the primary mode of failure, with overheating of the weakened thinned tubes subsequently leading to escalation. For a PFR evaporator, if sodium flow was maintained and the incident was detected within a few seconds of an escalation to 100 g/s leak, a major escalation of the event would be unlikely if the steam side was depressurised rapidly after leak detection.

The relative importance of wastage and overheating in a European Fast Reactor configuration of SGU, using Mod 9Cr1Mo steel tubes, was also investigated. The objective was to determine what leak detection targets should be set to avoid economic write-off of the SGU. These tests showed that wastage was the principal escalation mechanism from an intermediate size leak to secondary tube failures. Overheating failures were possible only if the leak raised the SGU pressure sufficiently to raise the boiling point of sodium to above 1100°C. It was concluded that the leak detection system needs to be able to detect and master a leak in the superheater region of 5 g/s or less within 10 s to prevent SGU write-off. This leak rate is unable to generate high temperatures around the boiling point of sodium and 10 s is a conservative minimum time to generate a secondary failure by wastage plus rupture of the wasted tube.

4.3.3 Destructive Examination of Materials

These examinations have been restricted to the secondary sodium circuit, since the funding available precluded the study of primary circuit components such as the Above Core Structure or Intermediate Heat Exchangers. The secondary circuit studies are in three parts:

i) Examination of carbon steel components exposed to sodium environments.

ii) Study of delayed reheat cracking in austenitic steel weldments.

iii) Study of secondary pipework transition welds.

Sections will be removed from certain secondary circuit components by end May 1995 and examination of these for cracking or damage will be completed by end March 1996.

Examination of Carbon Steel Components

Following cracking of 15Mo-3 steel in the Superphénix storage drum and Kalkar sodium storage vessels, questions remain over the use of as-welded carbon steels for structures in contact with sodium or sodium vapour. Examination of suitable as-welded joints in the PFR carbon steel tank farm vessels which have been exposed to sodium vapour is being carried out to provide assurance that similar steels (A42 and A48) likely to be used for EFR roof structures would not be susceptible to the type of cracking observed in 15Mo-3 steel. Carbon steel could then be used in the as-welded state for roof constructions, avoiding the need for costly stress relief heat treatment.

Study of Delayed Reheat Cracking

Delayed Reheat Cracking (DRC) was the main mechanism responsible for the cracking problem at 500°C to 520°C in the weldments of the PFR reheater and superheater shells. This is also the case for cracking in secondary pipework weldments in Phénix and it was a major contributory factor in two cracking incidents in PFR secondary pipework which contained notch-like features. All of these incidents have involved the Ti-stabilised Type 321 stainless steel.

It is accepted from the literature that Type 316 steel, and therefore by inference also the EFR Type 316(L)N steel, is less susceptible to DRC than the Ti-stabilised Type 321 steel of PFR, however, the data base of experience of DRC on Type 321 steel is itself meagre. In particular there is little information on section thickness effects and at relatively low temperatures (≤600°C). Suitable welds chosen for examination in PFR are those of the secondary pipework and the reheater/superheater vessel shells. These will allow examination of cracking, if any, in both good geometry and poor geometry welds and whether cracking is likely to occur in thin sections. Selected welds will be sectioned to look for cracking caused by DRC.

Study of Transition Welds

At present, there is still a need to define suitable design rules for transition welds. Such welds will be required between the roof and the primary vessel and between the secondary sodium pipework and the steam generators of future fast reactors. There is a history of problems with transition welds in conventional power stations but this has been with thick section welds operating at typically 600°C, where creep-related failures have occurred in a decarburised zone in the ferritic steel.

Transition welds in PFR which operate at high temperature are to be examined for microstructural changes. These are between the $2^1/_4$Cr1Mo steel evaporators and the Type 321 secondary pipework steel, using Inconel filler, and operate at or below 490°C. Examination of a transition weld for damage or microstructural changes will provide important information for future fast reactors, although the ferritic steel is different from that proposed for EFR (Mod 9Cr1Mo steel) and the temperature of the transition welds in EFR may be higher.

5. UK Activities in Support of the CAPRA Project

5.1 Introduction

The CAPRA project is aimed at examining the use of fast reactors to consume plutonium. It was set up by the French CEA as part of an R&D programme on the management of radioactive waste. Since September 1993 the UK has participated in it, under the leadership of BNFL, with the objective of continuing an interest in developments in the field of fast reactors, and of exploring the options available for the use of plutonium.

The main activity of the project in this period has been to examine the feasibility of a Pu-consuming core compatible with the EFR primary circuit design. It uses mixed-oxide fuel with high Pu enrichment, but compatible with current fabrication and reprocessing technology. The UK work was mainly theoretical, in the areas of reactor physics and performance, reactor safety, and fuel performance, and was done by AEA Technology and NNC personnel. It is reported in the CAPRA Feasibility Report, and in papers to the two CAPRA seminars, at Cadarache in March 1994 and Karlsruhe in September 1994.

5.2 Reactor Physics

The reference CAPRA oxide core is characterised by high Pu concentration, and by radial heterogeneity. The reactor physics calculation methods for such cores have been examined and a calculation route using the ERANOS code system has been set up.

Two methods of controlling the reactivity of a core with high Pu concentration are available: either it can be diluted as compared with a conventional core, or neutron-absorbing poisons can be incorporated. An initial study showed that the safety parameters of the poisoned version are not good, so further study concentrated on the dilute version.

The dilution of the core gives valuable flexibility. Studies were undertaken to examine the use of this core to accommodate Pu with a range of isotopic compositions, varying from "high-quality" Pu typical of that generated by a PWR without recycle to "poor" Pu which has been recycled several times in a CAPRA core, and to investigate the possibility of increasing the fuel burnup substantially.

In all cases the core performance was determined in terms of safety parameters such as the Doppler coefficient and the sodium void reactivity, the fuel cycle length (taking account of the rapid loss of reactivity with burnup), the absorber worth requirement, the fuel residence time, etc. It was possible to demonstrate the feasibility of the core, including the variants mentioned above, in terms of overall safety and economic performance.

5.3 Reactor Safety

The performance of the core under accident conditions, both within the design basis and beyond it, was examined. Within the design basis the protective system will operate in a satisfactory manner and the performance is similar to or slightly better than that of a conventional core. In accidents beyond the design basis, where the protective system is assumed to be defective, analysis shows satisfactory performance, but there is a need for validatory experimental data to confirm the performance of the fuel and diluent material.

5.4 Fuel Performance

The CAPRA mixed oxide fuel has high Pu enrichment, up to 45%, and would be irradiated to high burnup. Codes to calculate the performance of the fuel during irradiation have been adapted to take account of CAPRA conditions. The features of most concern are the migration of Pu and the chemical composition of the fuel material. The predictions have been compared with PIE data from a range of sources. The comparisons have been used to adjust the calculation parameters and validate the codes.

5.5 Related Activities

UK attaches have been active throughout the period at Cadarache, contributing to the programme of reactor physics validatory experiments on the MASURCA zero-energy assembly. UK attaches have also contributed to the commissioning team on the MONJU reactor in Japan.

UK delegates have contributed to the IWGFR Technical Committee Meeting on Material-Coolant Interactions, at O-Arai in Japan in June 1994, and to the International Safety Conference at Obninsk in Russia in October 1994.

A REVIEW OF FAST REACTOR ACTIVITIES IN BRAZIL

A. MENEZES
Advanced Studies Institute-
Aerospace Technical Center,
São Paulo, Brazil

Presented by Xu Mi

Abstract

The Instituto de Estudos Avançados is coordinating a project which aims to establish a basic know-how for an eventual utilization of fast reactors in Brazil. In spite of the many limitations due to reduced funds, we firmly believe that activities in this field should be maintained if we wish to keep fast reactors as an option for future use. A reference design for a 60 MWt experimental reactor has been prepared, based on some parameters and general description of fast reactors built or being designed in other countries. Also, research activities in areas such as metallic fuel, materials, etc. have been started.

1- Introduction

Brazil is a large country, with a population of a little more than 150 million distributed on its 8,5 million square kilometers. The "per capita" electricity consumption is of the order of 1500 kWh, a value which is low when compared to most advanced countries. Around 95% of the electricity come from hydric resources, most of them located in the southeast region of the country. The estimated hydroelectric potential is 260 GW, from which around 60 GW is being used. A recent study made by the Brazilian Federal Energy Board -ELETROBRÁS-, which takes into consideration different scenarios for economic growth, the resources available today and also the possibility of exploiting new resources, indicates that by the year 2015 the hydroelectric potential will be exhausted and, from that time on, the demand will have to be supplied by a growing number of thermoelectric power plants, either conventional or nuclear.

Brazil has presently one reactor (PWR/Westinghouse, 626 MWe) in operation and two others (PWR/Siemens, 1300 MWe) are still waiting for a government decision as to their construction. In fact, this decision has been postponed for several times as a result of the costs involved. Only recently the Brazilian economy is giving signs of stabilization, after a long period of very high inflation rates. Also, both internal and external national debts resulted in a reduction of the investments in nuclear power Research and Development (R&D) activities. In spite of the tight budgets, some research institutions are directing their best efforts towards preserving the nuclear option alive for a future use. This is the case of the Instituto de Estudos Avançados-IEAv (Advanced Studies Institute), which has been involved with fast reactor research since 1979, even though not continuously.

2- Historical Background

In 1969, the Instituto de Engenharia Nuclear-IEN (Nuclear Engineering Institute) initiated its activities related to fast reactors. As part of a contract with TECHNICATOME (France), a thermal-fast reactor has been designed, but it was not constructed due to problems with fuel supply. In 1972, a small sodium loop (100 KW) was inaugurated for studying heat transfer and several aspects of sodium technology.

In 1981 a decision to intensify the activities in fast reactors resulted in a cooperation agreement with ENEA (Italy), mainly for training in several areas. Also a contract with ANSALDO/NIRA (Italy) was signed to design and construct three modern sodium loops and auxiliary systems. Due to insufficient funding the loops were never constructed.

` In 1985, a new technical cooperation agreement was signed, now with Argentina, for a joint work in fast reactors Many discussions took place, but nothing has been done

Fast reactor activities at IEAv initiated in 1979, when it has been created a technical group with the objective of doing research on the utilization of nuclear reactors for space applications, with the main focus on neutronics and thermohydraulics That work (which did not last long!) gave the group a better understanding of the importance of fast reactors for electricity generation for civilian applications It was then decided to re-direct the work towards investigating different fast reactor core and blanket configurations for efficient utilization of Thorium, due to its large resources in Brazil Several ideas came up and a number of technical papers have been published In 1988, fast reactor activities at IEAv were discontinued because of the many difficulties in having the participation of other research institutions in Brazil, mostly involved with thermal reactor work

The main reasons for all these setbacks can be associated to the following a) the initiatives were isolated, with no search for institutional cooperation, b) the country has been, for a long time, going through economic problems and, c) the public opposition to nuclear energy was high These factors, and maybe some others, have never stimulated the "decision makers" to take seriously a long term and certainly costly R&D fast reactor program

In 1992, after many discussions, we succeeded in convincing the authorities that a long term fast reactor program should be maintained, if the country wished to keep this reactor concept as an alternative for a future use A R&D project was then established with the objetive of having, within 25-30 years, an experimental reactor in operation, in which relevant experiments could be performed Cooperation among institutions is to be a key factor

Since 1992 we have been working in what we call Feasibility Phase, which has, among others, the objective of submitting to the Brazilian authorities, a report composed of six volumes 1) Global View; 2) Reference Design (primary circuit) for an Experimental Fast Reactor, 3) Survey of Brazilian Research Institutions, 4) Survey of Brazilian Industrial Park, 5) Survey of Brazilian Universities and, 6) Preliminary Planning for Next Phases These volumes are planned to be ready by the end of 1995 and will be useful for identifying possible partnerships

3- Managerial Strategies

The "environment" which prevailed at the beginning of the Feasibility Phase was characterized by tight budgets and also by a strong disbelief from a large portion of the Brazilian nuclear community with respect to big projects In order to minimize the chances of a new setback, we decided to establish the following group of strategies

- **For motivating the internal technical team:** the project is to be conducted in phases of duration no longer than 3-4 years, with the objectives for each phase being defined in such way as to create a feeling that they can be reached, even with the present economical difficulties

- **For motivating the participation of other research institutions:** the approximation with the institutions must emphasize the several technical challenges in areas such as reactor safety, fuel, materials and so on, and to avoid discussing merits of fast reactors with a community biased towards thermal reactors

- **For motivating the "decision makers":** each phase of the project must generate the maximum number of results which are not only relevant, but also which are easy to be evaluated by the "decision makers". In practice, this means that every result should be transformed in an asset for larger funding The lesson we have learned from our past experience is that we should qualify for "big money" after demonstrating that good results can be produced with "small money"

- **For promoting the importance of fast reactor R&D activities in Brazil:** the introductory section of all technical papers and articles should give a clear indication that whatever is to be shown or discussed, represent a contribution to the long term objective of having a fast reactor basic know-how well established in Brazil.

4- Present Status

Research activities are being (or planned to be) executed in a few areas:

1. U-Zr alloys are being studied in cooperation with the Instituto de Pesquisas Energéticas e Nucleares-IPEN (Nuclear Research Institute), with the objective of learning how to fabricate and characterize them, in laboratorial scale;
2. Also with the same Institute, metal fuel recycling is being researched using electrorefining techniques. We expect to have a first sample of actinide electrodeposited by the end of 1995;
3. A laboratorial scale continuous current electromagnetic pump has been developed and tested at IEAv, using mercury as working fluid. Work is continuing towards the development of a small scale alternating current pump.
4. Activities related to HT-9 ferritic steel have been initiated in cooperation with IPEN and with the Instituto de Pesquisas Tecnológicas-IPT (Technological Research Institute). The first sample of this material produced in Brazil has been obtained at IPT. At this moment the first sample is being characterized.
5. A small thermohydraulic nuclear laboratory was set within a collaboration between IEAv and the Instituto Tecnológico de Aeronáutica-ITA (Aeronautic Technological Institute), in order to perform experiments related to natural convection phenomena. ITA is an engineering school, which implies that beyond the technological research aspects, also the educational ones are involved.
6. A Reference Design for the primary circuit of a 60 MWt experimental fast reactor is completed. Some parameters and characteristics have been selected mostly based on the Integral Fast Reactor (IFR) concept. Fuel pin dimensions and other data were used for calculations which lead to a general core configuration. Some calculations were also independently performed at IEN, and the differences in methodologies are being evaluated. Table 1 shows the main primary circuit parameters.
7. Research on sodium-related corrosion is under discussion to be done in collaboration with IEN.

5- Conclusions

In our understanding, the achievement of important technical results should be done simultaneously with a step-by-step dissemination of a “fast reactor culture” in Brazil. So far, we have been successful in getting the cooperation of excellent technical groups, from different research institutions, by proposing their involvement in activities which may represent a challenge to their technical expertise. We are aware that there is still a lot to be done and also a lot to be learned, mainly from the most advanced countries. It is our feeling that the continuation of fast reactor activities in Brazil depends upon the strengthening of the cooperative work already initiated, and also on the generation of results which are easy to be evaluated by our “decision makers”.

Following the Feasibility Phase, which ends in December, 1995 we intend to increase the level of details of the Reference Design. Considering that an operating experimental reactor is still distant in the future, for all purposes the Reference Design is our “experimental installation” and is to be used as the basis for calculations related to safety analysis, neutronics, thermohydraulics and so on.

Finally, it must be emphasized that even small programs from developing countries may give an important contribution to the fast reactor field, mainly through intellectual work and also through performing a few experiments. Our objective is to establish a basic know-how which can be useful for future use in Brazil. The interaction with the more advanced programs can serve as a guide, allowing us to reach a degree of maturity which will result in benefits both to Brazil and also to the international fast reactor community.

TABLE 1. MAIN PRIMARY CIRCUIT PARAMETERS

Core	
Thermal Power (MW)	60,0
Electric Power (expected) (MW)	20,0
Core inlet temperature (C)	370,0
Core outlet temperature (C)	470,0
Fuel type	U-10%Zr
Fuel loading (kg U)	811,0
Volume fractions (%):	
Fuel	41,0
Sodium	37,7
Structure	21,3
Sodium void reactivity (Δk)	-0,0327
Active core height (cm)	62,0
Expected core average burnup (MWd/kg)	70,0
Maximum central fast flux ($n/s/cm^2$)	$\approx 2x10^{15}$
Heat transfer system	
Number of pumps	2
Operating primary flow rate (kg/s)	452,0
Nominal operating pressure (MPa)	0,11
Sodium volume (l)	

SWISS FAST REACTOR RESEARCH PROGRAM

R. BROGLI
Paul Scherrer Institute,
Villigen, Switzerland

Abstract

The motivation for the small, Swiss research effort on fast reactors is to maintain a knowledge on the safety problems of the fast reactor in particular on Superphénix. In the western part of Switzerland, mostly in region of Geneva, there is a widespread fear of being harmed by a possible accident at Creys-Malville situated about 100 km downstream the Rhone river but upwind the prevailing westerly winds from Geneva. The objective information gained through the research effort and the information exchanges with the French Safety Authorities allow the Swiss authorities to evaluate the risks and to put them into perspectives with the risks experienced from other sources and the limiting values of the Swiss legislation.

There are three research activities, one in reactor physics trying to learn on various influences on the reactivity a second in thermohydraulics studying the mixing processes of sodium at fluid interfaces and a third on fuel materials for potential use in future plutonium burning fast reactors.

2 Reactor Physics

Over the years the PSI fast reactor physicists have become specialists in testing and validation of nuclear cross-section used for the calculation of reactivity and reactivity-coefficients based of benchmark-problems and experiments. One can certainly claim that for the usual fast and thermal reactor core arrangement the nuclear cross-section and the methods are adequate. However for the envisaged new core arrangements and compositions to burn actinides instead of producing them the methods and nuclear cross-section have to be re-examined, since certain isotopes which played a minor role in the past are gaining importance.

2.1 NEA Benchmark

The 600 MWe high plutonium enriched, steel reflected NEA-Benchmark was evaluated by several institutions. Multiplicationsfactors, Sodium-Void-Coefficient, Doppler-Coefficient and Burnup-Defect were compared. As the table in figure 1 is showing the agreement on the reactivities is not satisfactory. Much worse however is the comparison on the Sodium-Void-Coefficients. The second table in figure 1 shows that a part of the difference is due to different calculated leakage effects.

2.2 Analysis of the PECORE-Experiments

In order to evaluate the effect of the influence of the nuclear data of the steel reflector on the leakage the PECORE Experiments were analysed using the JEF-2.2 Data Base. The various cross-sections of iron were then artificially doubled in order to see their influence on the reactivity and on the reaction-rates of the whole system. It is obvious from figure 2a that the Elastic-Scattering cross-section has an significant influence on the reactivity. Furthermore the iron cross-sections and only the iron cross-sections were replaced by iron cross-section from other data bases. The results are

Institute	K_{eff}	k_∞	Absorption (%)	Leackage (%)
ANL	1.107	1.233	89.8	10.2
CEN Cadarache	1.112	–	–	–
PNC	1.125	1.230	91.5	8.5
Toshiba	1.135	–	–	–
PSI	**1.128**	**1.226**	**92.0**	**8.0**
IPPE	1.115	1.260	88.5	11.5

Multiplicationfactor k_{eff}, k_∞ and neutron-balace for fresh fuel

Institute	ANL	CEN Cadarache	PNC	PSI
Inner core without leakage	+ 2785 – 1235	+ 2484 – 1540	+ 2310 – 1400	+ 1994 – 1318
Inner core Total reactor	+ 1490 + 1550	+ 1110 + 944	+ 1450 + 910	+ 950 + 676

Sodium-void-coefficient for fresh fuel

FIG. 1. NEA-Benchmark.

k_{eff}	k_{eff}-Variation (in % of Δ k/kk')	Modified Cross-Section of Iron	C8/F5 (E-C)/C	F8/F5 (E-C)/C	F9/F5 (E-C)/C
Experimental Standard-Deviation			± 0.0195	± 0.0162	± 0.0114
1.001	0.	Reference calculation	+ 0.0236	− 0.0236	+ 0.0158
1.043	+ 3.09	elastic	+ 0.0179	− 0.0037	+ 0.0210
1.013	+ 0.21	inelastic	+ 0.0228	− 0.0208	+ 0.0165
1.008	− 0.28	absorption	+ 0.0244	− 0.0265	+ 0.0152

Global sensitiveness of the multiplication factor k_{eff} and the ratio between the calculated and measured ratio of reaction ((E-C)/C) in the core centre

FIG. 2a. PECORE- experiments.

k_{eff}	k_{eff}-Variation (in % of Δ k/kk')	Modified Cross-Section of Iron	C8/F5 (E-C)/C	F8/F5 (E-C)/C	F9/F5 (E-C)/C
1.024	+ 1.27	JENDL-3.1	+ 0.0207	− 0.0113	+ 0.0184
1.001	0.	JEF-2.2	+ 0.0236	− 0.0237	+ 0.0158
1.009	− 0.16	ENDF/B-VI (Rev.2)	+ 0.0241	− 0.0254	+ 0.0154
1.007	− 0.34	BROND-2.1	+ 0.0245	− 0.0272	+ 0.0151

The multiplication factor k_{eff} and the ratio between the calculated and measured ration of reaction ((E-C)/C) in the core centre resulted from calculations with various data of ^{56}Fe in the radial reflector

FIG. 2b. PECORE-experiments.

seen in figure 2b. The variations in the reactivities are again significant, indicating that inconsistencies exist among the iron cross-section of the various data bases and that these differences have a significant impact on the reactivities.

3 Thermohydraulic

The present LMFBR-related thermal-hydraulics research programme consists of experimental and analytical investigations of the effects of stratification and thermal conductivity on the interaction (mixing) between two horizontal fluid layers of different velocities and temperatures. This subject has been chosen because of its suitability with respect to available experimental facilities and expertise, its fundamental character and its relevance to passive decay-heat removal in pool-type LMFBRs.

Mixing layers, which occur in many engineering applications, are shear layers between fluid streams of different velocities and are characterised by transfer of mass, momentum and energy due to vortex formation. For mixing layers with density stratification, buoyancy forces are important besides shear stress and influence considerably the intensity of mixing. Stratification has been recognised to be of importance in heat exchange between sodium streams of different temperature in the upper plenum of FBRs under decay-heat-removal conditions.

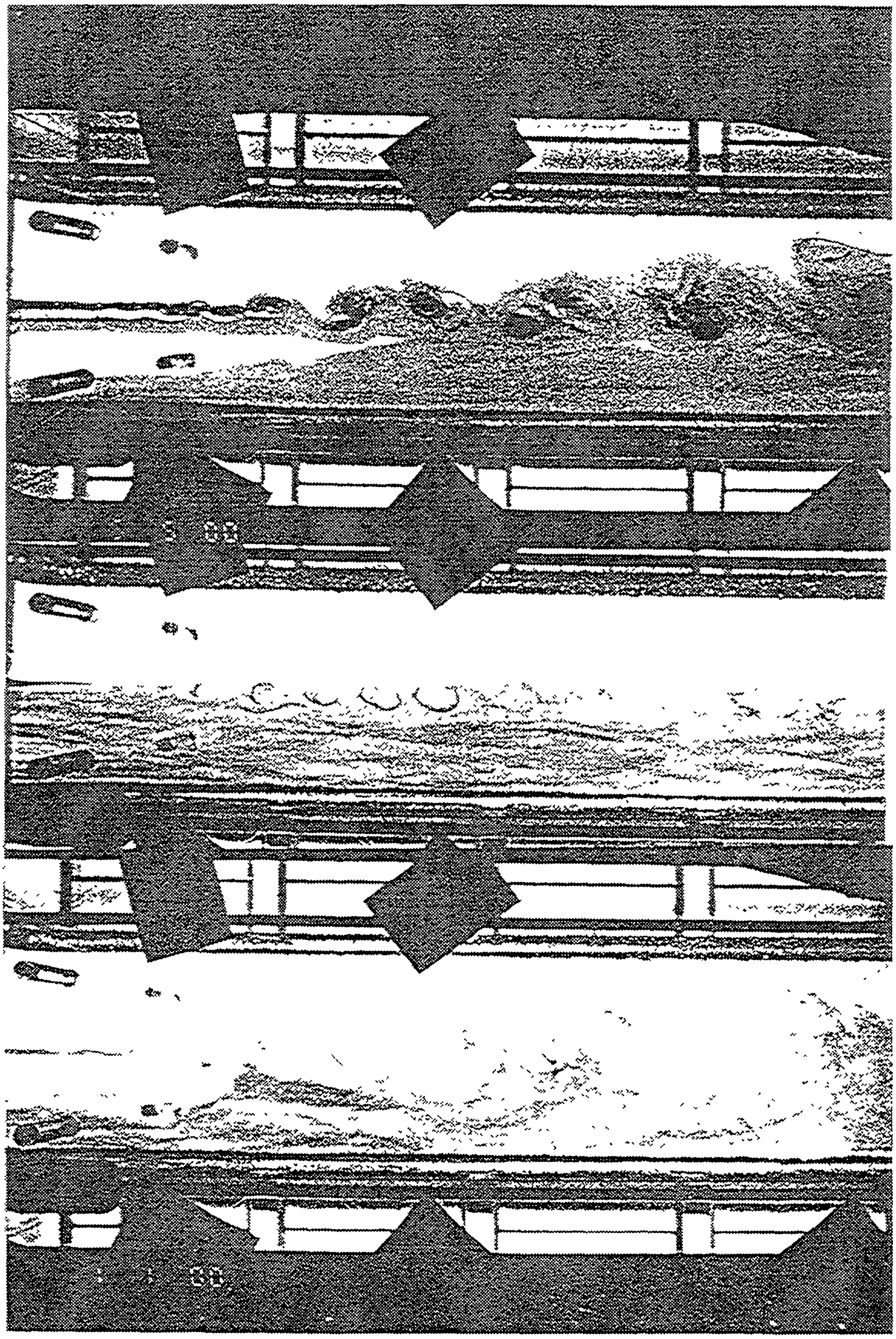
The basic physical phenomena in mixing layers have been thoroughly investigated experimentally and theoretically, in the past. In spite of knowledge acquired in extensive experiments with air and water, questions concerning a more detailed understanding of mixing-layer phenomena and particularly the influence of the relatively high heat conductivity of liquid metals (very low Prandtl number), still exist. A satisfactory validation of code predictions and further development of turbulence models must also be performed. Thus, in order to obtain sufficient and reliable experimental data for the modelling of characteristic phenomena, additional appropriately designed experiments are needed.

The main purpose of the Swiss mixing-layer research programme is to investigate the thermal-hydraulic phenomena in horizontal shear layers with particular attention paid to the effects of Richardson (Ri) and Prandtl (Pr) number. Therefore, experiments are to be performed over a wide range of velocity and temperature differences between the two streams and with water and sodium. In connection with safety considerations for pool-type LMFBRs, a reliable experimental data base should allow one to validate codes used to calculate the flow fields in the pools.

3.1 WAMIX Experiment

Flow visualisation is one of the main objectives in the WAMIX test-section made of acrylic glass. In the initial phase of the tests, visual observations lead to a qualitative understanding of the developing shear layer.

The test-section, consisting of two inlet chambers, equipped with honeycombs, grids and contractions, provide almost laminar, parallel streams with different velocities, which merge after the wedge and form a mixing layer. Density differences of the two streams, brought about by unequal temperatures, create buoyancy forces which support or oppose the shear forces, depending on the sign of the density gradient (see figure 3).



a)

b)

c)

FIG. 3. Effect of thermal stratification
a) neutral (Ri = 0) b) stable (Ri > 0) c) unstable (Ri < 0).

A considerable effort was devoted to the testing of different experimental and measurement techniques. The generation and interaction of streamwise vortices and the effect of stratification on streamwise vortices were visualised with the laser sheet technique, using luminescent dye diluted in water. In order to investigate the formation of streamwise vortices (visualisation by a moving laser sheet) a conveyer system was used (see figure 4).

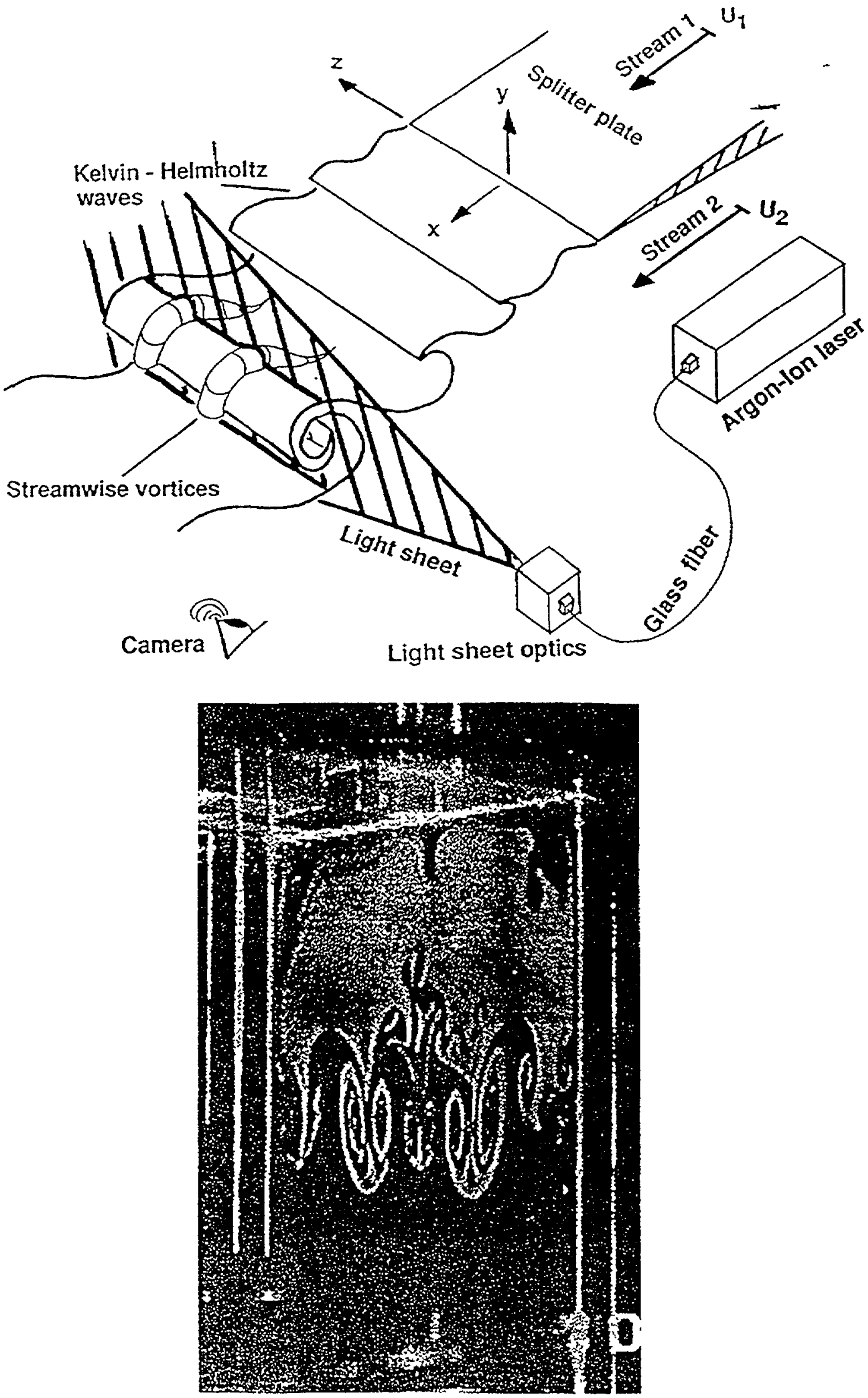


FIG. 4. Visualization of streamwise vortices by laser light sheet technique.

3.2 NAMIX experiment

Originally the same geometry as for WAMIX was planned to be used for the NAMIX experiment. Then, modifications of the test-section which turned out to be necessary because of experimental requirements lead to design difficulties and would have caused too-great expenses. Therefore, it was decided to change the design completely and to build a test-section with homogeneously- turbulent inlet flows. Presently, this test-section is in the final design status (preparation of technical drawings).

3.3 Analytical work

The experiments conducted in this programme are designed to provide information for the validation of general-purpose fluid dynamics codes (like FLOW3Dor ASTEC, and of direct numerical simulation codes (such as FLOW-SB).

Numerical work at PSI, using the available codes ASTEC and FLOW3D has the main objective to check the ability of such codes to reproduce (with reasonable accuracy) the overall development of stratified mixing layers at various Prandtl numbers.

Up to now, most of the work has been carried out with the code ASTEC. After setting up of a suitable mesh for the numerical simulation of the experiment, calculations of velocity profiles in the flow conditioning of the test-section and an investigation of the effect of channel wall inclination on mixing-layer development have been carried out.

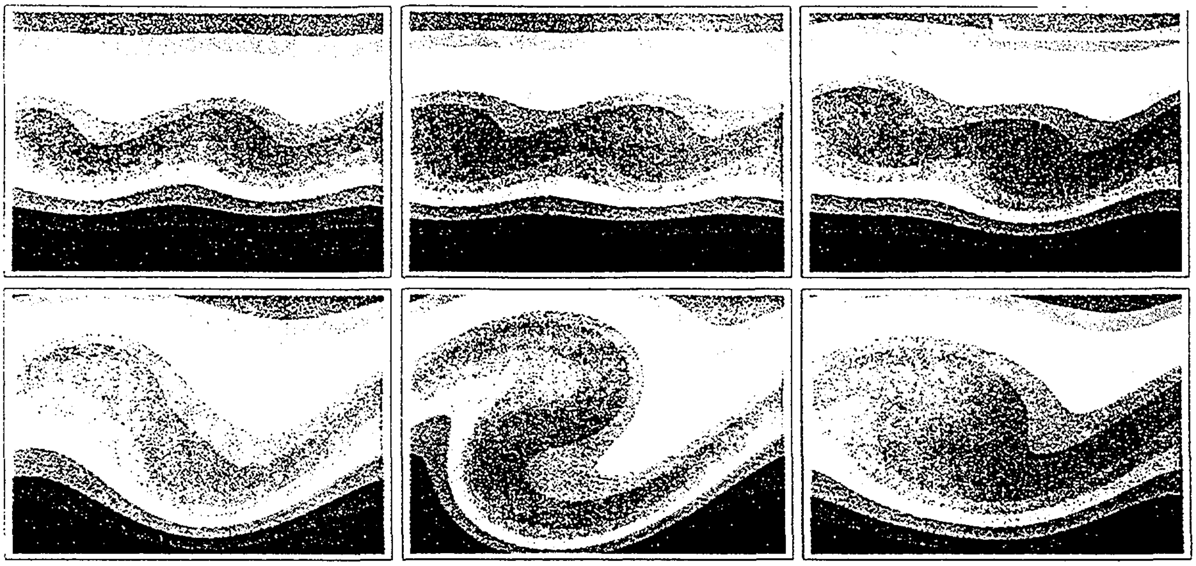
In 1993, calculations with FLOW3D were also started to be included in the validation procedure. Recent work performed for a benchmark exercise, organised by the IAHR Working Group on Advanced Nuclear Reactors Thermal Hydraulics, concerns the numerical simulation of vertical buoyant jets.

A survey of analytical work was presented at the international "Computational Fluid Dynamic Services (CFDS)" conference. The results of some calculations were also included in a contribution to NURETH-6 and the benchmark calculation results are submitted to the next IAHR Meeting.

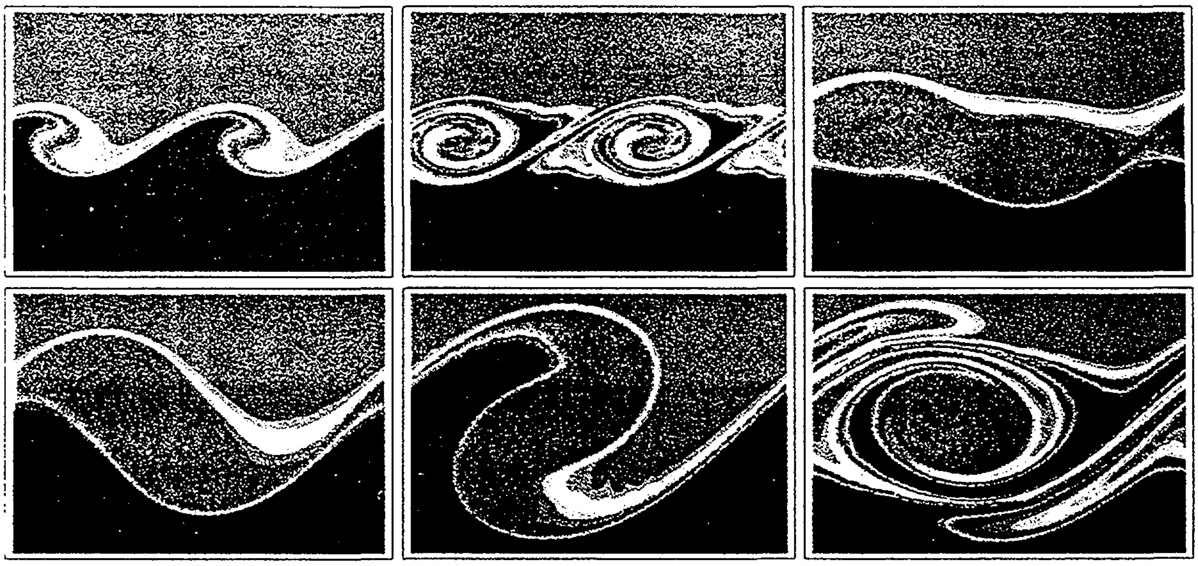
For the purpose of direct numerical simulation (DNS), the pseudo-spectral code FLOW-SB (originally developed to study wall turbulence) has been adapted to the temporal (spatially periodic), stratified mixing-layer problem.

Very good progress has already been achieved with respect to the development of this code. Many modifications have been successfully implemented and a number of interesting results on mixing-layer development with 2D and 3D models have been obtained.

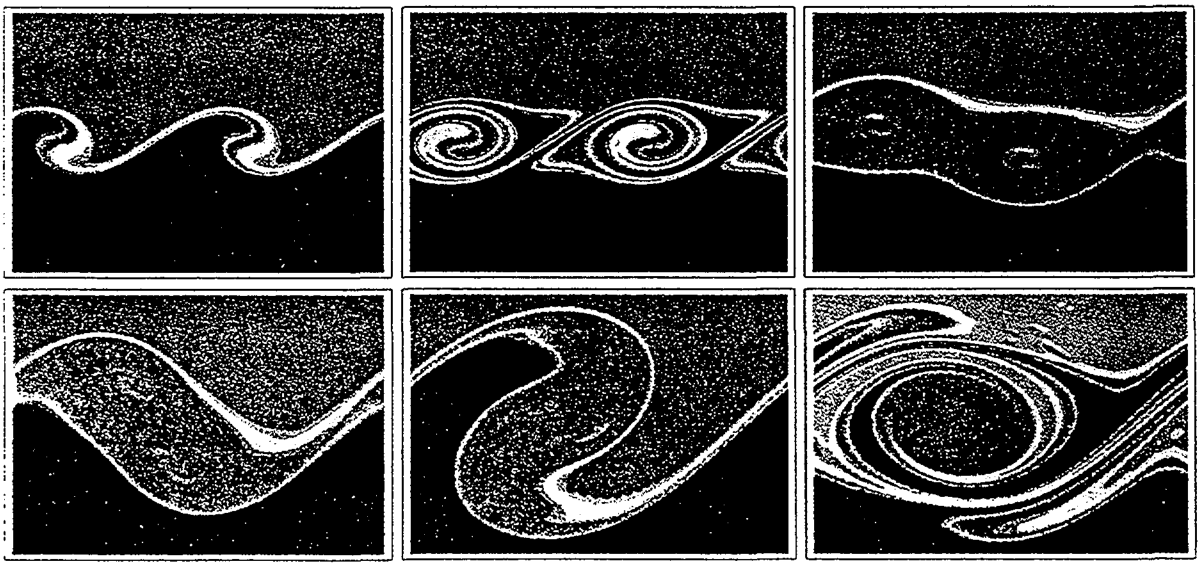
Recently, an analysis of stratified mixing layers at different Ri and Pr numbers has been performed. With the help of mathematical models for visualising flow structures and temperature fields a direct comparison of simulation results, obtained for various Prandtl and Reynolds numbers, is possible. The drastic effect of Pr on the temperature distribution in a neutral mixing layer is demonstrated in figure 5.



(a) Liquid sodium (350°C, Pr=0.00535)



(b) Air (20°C, Pr=0.69)



(c) Water (80°C, Pr=2.2)

FIG. 5. Temperature distribution with time for various fluids.

4 Material Program

In the framework of the CAPRA project the fuel option for amplified plutonium consumption is being studied (figure 6). Coprecipitation of uranium and plutonium by a sol-gel process offers advantages in the fabrication and possibly also in material characteristic over the classical power - pellet fabrication process. The dissolution characteristics of coprecipitated uranium plutonium oxides with concentrations up to 55 at% need to be clarified. Small amounts of material have been prepared and detailed characterisations are in progress. With the preparation of larger quantities (100g), the evaluation of a modified dissolution test and the verification of fabrication conditions will be aimed. For the preparation of inert matrices with actinides, the preparation conditions and material characteristics relevant for in-pile testing are to be evaluated. Based on discussions with the CAPRA team and in agreement with CEA/DEC PSI concentrates its effort on promising candidates such as the oxide system CeO_{2+x}-PuO2 with some additions of UO_2. The pelletisation of nitrides and oxides with the existing pellet press is made available for the preparation of irradiation specimens. For the irradiation experiment MATINA-1 in the Phenix reactor titanium nitride pellets have been fabricated at PSI by compacting titanium nitride powder and sintering the compacts at 1750°C. Fabrications tests are under way for the nitride system of zirconium and uranium with the goal of preparing zirconium americium nitride as an inert matrix for the incineration of americium at a later stage.

The layout of a CerMet (ceramic-metal) fuel based on ceramic particles as an advanced solution for plutonium incineration without uranium is addressed with model calculations for thermal and mechanical behaviour.

- Preparation experiments and solubility tests for $(U_{1-x}Pu_x)O_2$ $(0.25 < x < 0.65)$ and for $(U_{1-x}Pu_x)N$ $(0.25 < x < 0.75)$ as possible materials for the efficient fission of Plutonium in a fast neutron flux (CEA/CAPRA)
- Fabrication of pure PuN-microspheres for CERMET fuel
- Design calculations for sphere-pac segments based on the idea of a CERMET fuel (CAPRA/TRABANT)
- Material preparation of (U,Zr)N and pelletization tests of TiN and (U,Zr)N for the irradiation experiment MATINA in the reactor PHENIX
- Experimental preparations of $(Ce,U)O_2$, $(Ce,U,Pu)O_2$ and $(Ce,Pu)O_2$ for the CAPRA core with lower Pu content
- Cleaning of Americium from waste streams of the Plutonium separation equipment (extraction chromatography)

FIG. 6. Materials for Actinide Transmutation.

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EUROPEAN UNION: REVIEW OF FAST REACTOR RELATED ACTIVITIES PERFORMED DURING 1994

W. BALZ, G. VAN GOETHEM
European Commission,
Brussels, Belgium

Abstract

The European Commission continued its fast reactor related activities on the same lines as in the past, in a reduced fashion however and with emphasis on applications to actinide and fission products transmutation.

1. INTRODUCTION

The European Commission continued its fast reactor related activities on the same lines as in the past, in a reduced fashion however and with emphasis on applications to actinide and fission products transmutation.

2. RESEARCH COORDINATION ACTIVITIES

The Fast Reactor Coordinating Committee (FRCC) pursued the exchange of information and the discussion of specific problems in the Safety Working Group (SWG), while monitoring benchmark exercises about the Russian BN-800 fast reactor in cooperation with IPPE/Obninsk through their subgroup WAC.

Regarding collaboration with Central Europe and CIS, the EC went on in 1994 with their assistance and cooperation Programmes PHARE and TACIS, which include also Fast Reactor research and development activities.

2.1 Safety Working Group (SWG)

At the 42-th FRCC meeting in October 1994, a report was presented about the past activities of the EFR ad-hoc safety club (AHSC), a group of senior experts from various countries (GB, F and FRG), who assessed the deterministic/probabilistic safety approach applied to EFR, aimed at making the risk of EFR lower or equal to that of the future LWRs. It was recalled that this activity led to a successful harmonisation of safety analyses among the EFR partners. Obviously the safety criteria of the SWG are more of a generic nature and could be improved by being systematically compared with those applied to EFR. Such European safety criteria for fast reactors could be applied also outside the European Union, e.g. to Russian fast reactors.

2.2 Whole core Accident Calculation group (WAC)

Unprotected Loss Of Flow (ULOF) comparative calculations of the Russian standard BN-800 core

As far as the WAC group is concerned, a final WAC report has been produced about the "Unprotected Loss Of Flow (ULOF) comparative calculations of the Russian standard BN-800 core", discussed at a WAC meeting at IPPE/Obninsk (9-12 November 1993) and presented at the International Conference on Fast Reactor Safety (3-7 October 1994, Obninsk). The conclusions can be summarized as follows:

- the Doppler effect seems to be the basic mechanism for going from super-critical to lower-than-supercritical conditions and fuel dispersal effects come up only a few milliseconds later, which then shuts down the reactor
- space-time kinetics are definitively needed when large materials motions appear, making the standard point-kinetics approach totally unappropriate

Steady state neutronics calculation of a nearly-zero void reactivity core design project of the Russian BN-800 fast reactor

In parallel to the above-mentioned activity of hypothetical accident calculations under severe transient conditions, a benchmark exercice was launched jointly by EC and IAEA for the "Steady state neutronics calculation of a nearly-zero void reactivity core design project of the Russian BN-800 fast reactor" of 2100 MWth. The idea was to investigate the capability of reducing to nearly zero the sodium voiding feedback reactivity of an axially heterogeneous fast breeder reactor when a sodium plenum is introduced above the core instead of the upper axial blanket. In addition to the sodium void reactivity calculation, it was decided to investigate a series of core neutronic performance characteristics, such as breeding ratios, burnup swing, peak linear power, power peaking factor, regional power fractions and control rod worths. Russia, Japan, India, Great Britain, USA and EFR participated. The EU supported the calculational effort of the participating EFR representative (SIEMENS) - see EUR Report 15133, dated 1994 and entitled "Fast Reactor with near-zero sodium void effect - Neutronics calculations for the IAEA/EC benchmark". IAEA supported the drafting of the final report (see IAEA Report TECDOC-731, dated January 1994 and entitled "Evaluation of benchmark calculations on a fast power reactor core with near zero sodium void effect").

Unprotected Loss Of Coolant (ULOF) comparative calculations of the Russian BN-800 core with nearly-zero void reactivity

As far as future activities are concerned, an extension of the ULOF benchmark exercise of the WAC Group has been launched along the lines of the above-mentioned nearly-zero void reactivity core concept of BN-800, now to be examined for severe transient accident conditions. At their meeting in May 94, the IWGFR of IAEA had agreed indeed to support this new BN-800 calculational exercise proposed by IPPE. Participation includes : Russia, Germany, France, UK and Italy as the traditional partners of the WAC comparative exercises, as well as USA, Japan and India as additional "IAEA" partners.

The main aim of this new comparative BN-800 calculation has been fixed at a EC sponsored meeting in Brussels on 11-12 July 1994. It consists of the following items:

- to establish a basis so as to evaluate the characteristics of a BN-800 type reactor with a nearly zero sodium void reactivity core design under hypothetical severe accident conditions, as far as an energetic ULOF is concerned
- to analyze the conditions which allow avoidance of prompt fuel and steel melting in the improved fast reactor core under ULOF-type and other severe accident conditions.

The accident under consideration is a LOF (Loss Of Flow) accident. It is proposed to analyze two cases :

- fully unprotected loss of flow (all safety rods are out of operation);
- partly unprotected loss of flow (all active safety rods are out of operation but all passive safety rods keep their ability to work normally);

IPPE provides the participants with the BN-800 input data for the core configuration. Decay-heat removal phase is not considered.

The main features of the reactor under consideration are chosen close to the BN-800 reactor with nearly-zero sodium void reactivity core design. In particular:
- core design (presence of the sodium layer and boron shielding above the core; the main subassembly thermal and hydraulics characteristics);
- the fuel type (mixed uranium plutonium dioxide);
- primary circuit layout; general design features of the core, pump and IHX;
- thermal and hydraulics parameters of the coolant for the primary and secondary circuits.

Reactor power is chosen in the range : 1500-2100 Mw (thermal)
Sodium void reactivity effect is (0 to -0.1)*10E-2
The number of the zones with different enrichment is 2 or 3.
A multi-batch loading scheme is assumed.

The calculational models should allow calculation of the fuel pin characterisation for the End of Equilibrium Cycle Core Loading Scheme and during the transient as well as the transient thermal and hydraulics processes in the reactor and in the primary circuit taking into account the sodium boiling. The models should include multichannel representations of the core (12-30 channels), with appropriate models of the out- and inlet plenum and the IHX, or pre-specified in-/outlet boundary conditions (pressure and temperature).

The following reactivity effects are taken into account :
- sodium thermal expansion and void reactivity;
- Doppler effect;
- axial core expansion;
- radial core expansion;
- control rod drive expansion.

IPPE provides reactivity worth tables and specifies correlations for calculations of structural feedback effects of reactivity (with reference data). The duration of the LOF scenario to be computed is supposed to be up to 1000s.

As a first step in this new cooperative effort, a "Consultancy meeting on IAEA/EC Comparative calculation for severe accident (ULOF) in a BN-800 reactor" was organised at IAEA, VIC Vienna, on 5-6 December '94, with the participation of FZK, CEA, AEA, IPPE, NUPEC and IAEA/IWGFR.

Several key items were discussed at that occasion, like:
1- sophisticated fuel pin mechanics modelling is needed to describe the long-term creep behaviour of the cladding during the expected long lasting boiling phase before pin rupture
2- sophisticated reactivity feedback models are needed because the low sodium void worth may generate long lasting near-critical situations without any dominant shutdown mechanism
3- the need to consider core meltdown situations for this BN-800 design (because of the requirement to develop appropriate accident mitigation techniques): even if the reactor is

obviously designed not to enter into such a degraded core situation, the scope of this benchmark is precisely to quantify the uncertainties and the residual risk of a beyond-design-basis accident (see conclusions of ad-hoc safety group in the EPR Project)
4- the importance of boiling/no melting scenarios and the validation of numerical tools, as well as the need for high-temperature and high strain-rate dependent materials databases.

A joint EC/IWGFR follow-up meeting is foreseen at EC, Brussels on July 26-28, 95 to discuss the first results of this new benchmark exercise with all participants.

3. RESEARCH PERFORMED AT THE JOINT RESEARCH CENTRE

Safety research is performed at the Safety Technology Institute (STI) of the Ispra Establishment and fuel research at the Transuranium Institute (TUI) of the Karlsruhe establishment, with emphasis however on transmutation processes.

3.1 Safety

Transmutation and European Accident Code (EAC-2) at JRC/STI-Ispra

In studies of Actinide and Fission Product transmutation interest has been focused on accelerator driven subcritical systems which are of particular interest if one considers fast actinide burners where the fissile isotopes of Neptunium, Americium, and Curium have a considerably smaller fraction of delayed neutron emitters (compared to the more common fuels U-238 and U-235), a small Doppler effect and a possibly positive coolant void coefficient. This poses a particular problem of control since the fraction of delayed neutrons is essential for the operation of a nuclear reactor in the critical state.

To overcome these problems various concepts of accelerator driven systems aiming at the transmutation of actinides and long lived fission products have been proposed in the recent past. The JRC presented a review of accelerator-driven sub-critical systems with emphasis on safety related power transients followed by a survey of thorium specific problems of chemistry, metallurgy, fuel fabrication and proliferation resistance.

Safety considerations of such new approaches and in particular the investigation of severe accident conditions are important. An investigation of a severe reactivity accident in an accelerator-driven fast reactor was earlier undertaken.
A new study deals with a Loss-of-Flow accident in which the primary pumps are assumed to coast down (e.g. due to a station blackout) in an accelerator-driven subcritical reactor. Different from the study above in which point kinetics was applied to a spatially uniform reactor in which only the Doppler feedback was considered, this study used the EAC-2 code for the multi-channel analysis of fast reactor accidents. The subcriticality was simulated by a programmed negative reactivity. To simulate the spallation neutrons coming from the target, a source was introduced into the point kinetics module. The reactor under consideration was the sodium-cooled 800 MWe fast reactor design used in the European WAC benchmark calculations of 1989. As a general tendency it was confirmed that an accelerator driven system reacts benignly to the introduction of a considerable positive reactivity such as introduced by sodium voiding. However, the flow coast down will nevertheless lead to core melting at a sizeable fraction of nominal power if the accelerator beam is <u>not</u> switched off. If the accelerator beam is switched off even as late as the early sodium boiling phase, the reactor power decreases rapidly and coolability of the reactor by

natural convection flow is likely. Therefore, investigations into passive means for shutting off an accelerator or for diverting its proton beam are of importance.

Advanced 3-D nodal methods for FR neutronics calculations

A doctoral study was completed about an improved method for solving the nodal 3D neutronics code HEXNOD which is the static part of the time-dependent code package HEXNODYN/. In the original HEXNOD code the 3D hex-z nodal calculation consisted in solving four 1D Sn calculations and combining them to obtain the neutron currents at the faces of a hexagonal cell. In the improved method the integral boundary sources method was used for solving the 1D equations. Moreover, higher order anisotropic moments order can be calculated and used for the combined solution. By using anisotropic moments up to the third order, a considerably increased accuracy was achieved for the reactivity calculation of an FZK SNEAK experiment simulating a disrupted reactor core. The new method is also similarly efficient as the old one when only the 0th and 1st moment are used. If one uses moments up to the third order the calculational efficiency will be about 30% lower.

3.2 Fuel Research/Advanced Fuel Development at JRC/TUI-Karlsruhe

3.2.1 Safety Aspects of Fuel Operation and Handling

POMPEI mixed nitrides and technetium irradiation experiment

The investigations of operational limits for future fuels were continued. The high burn-up irradiation experiment POMPEI (POM PEtten Irradiation) in HFR-Petten has achieved the goal burn-up (27.8 ${}^{a}/_{0}$) and was unloaded from the reactor for post irradiation examination (PIE). The high burn-up irradiation experiment POMPEI, containing targets loaded with mixed nitride fuel and technetium has reached its goal burn-up after 270 days of irradiation in HFR Petten. The aim of this experiment is to obtain data on the evolution of structure, fission products and the chemical behaviour at high burn-up.

3.2.2 Work in Relation to Minor Actinide Transmutation

Fabrication of fuel pins for the HFR irradiation experiment TRABANT (TRAnsmutation and Burning of ActiNides in TRIOX)

In the frame of the project CAPRA (Consommation Accrue de Plutonium dans les RApides), irradiation experiments are planned to study the transmutation of actinides and long-lived products and the behaviour of the new fuel types, which are either defined by a very high content of plutonium (> 40 at%), uranium and minor actinides (MA) or by the complete absence of uranium (in order to avoid additional breeding).
The irradiation experiment TRABANT is planned and executed in a trilateral cooperation with CEA and FZK. The Institute activities are concentrated on the fabrication of fuel pins to be irradiated in a TRIOX capsule in the HFR reactor at Petten. The purpose of the irradiation is to study the behaviour of the new materials under neutron irradiation in order to get information on such parameters as Pu- and MA-distribution, separation processes and solubility behaviour in nitric acid.

Experimental Feasibility of Targets for TRAnsmutation (EFTTRA)

In the field of research on nuclear waste management, the possibility of separating and transmuting the long-lived radioactive nuclides, with the aim of reducing the radiotoxicity of the final waste, is being investigated. In order to contribute efficiently to the development of materials for the transmutation, the Institute is collaborating with CEA (France), ECN (The Netherlands), EDF (France), and FZK (Germany), with the aim of setting up joint experiments. The group was named "Experimental Feasibility of Targets for TRAnsmutation" (EFTTRA).

The goal of the EFFTRA collaboration is the study of materials for transmutation, including the fabrication and characterisation of fuels and samples, their irradiation, and test of their in-pile behaviour. The work should be limited to the basic study of fundamental aspects of the problem. Being the subject of other programmes, the reprocessing and the partitioning as such are not considered in EFFTRA, but their interrelation with transmutation should of course be taken into consideration. This applies to strategies, where the type of reactor to be used for transmutations is, among others, an important parameter.

It has been decided to focus efforts on the study of materials for the transmutation of ^{99}Tc (metal), of ^{129}I (compound), and of Am (in an inert matrix); the homogeneous recycling of Am is the field of other international collaborative efforts, as illustrated for example by the irradiation experiments SUPERFACT 1 and SUPERFACT 2 (in preparation). The first phase of the EFTTRA collaboration was defined as the irradiation of Tc samples, ^{129}I compounds, and "empty" (without Am, but partially using U to simulate Am) inert matrices, and the related post-irradiation examinations.

Radiation Dose Aspects of Fuels used for the Transmutation of Minor Actinides (MA)

Nuclear waste, either in the form of spent fuel reprocessing, is associated with a radiotoxicity potential due to minor actinides (MA) and fission products (FP). The possibility of partitioning minor actinides out of the waste and transmuting them into less hazardous nuclides has been proposed and its technical feasibility is being studied.
Several concepts of MA transmutation, relying on existing nuclear power stations, are being studied theoretically and experimentally. In one, the recycling of MA with plutonium in the "self-generated mode" is being studied either in thermal (PWR) or fast reactors (FR). Another concept, the transmutation of minor actinides in a fast reactor has been investigated in practise in the irradiation in the fast reactor PHENIX.
Radiation dose levels have to be calculated at fabrication and discharge for the different recycling schemes. In order to assess the accuracy of these predictions, measured radiation doses of minor-actinide-containing fuel irradiated during, e.g. the SUPERFACT programme, can then be compared with those calculated.

The following nuclear fuel cycles have been considered for the self-generated recycling of Pu and minor actinides

- PWR and Fast Reactor reference fuel cycles (R1)
- PWR self-generated transuranium (TU) recycle (R2)
- FR self-generated transuranium recycle (R3)

As a case study, minor actinide recycling in a Fast Reactor has been studied on a series of oxide fuels containing ^{237}Np and ^{241}Am at low and high concentrations (experiment

SUPERFACT). Four types of mixed actinide oxide fuels were prepared at the Institute in accordance with the homogeneous fuel concepts (SF14,SF15). The fuels have been irradiated in the PHENIX power station and are currently undergoing post-irradiation examinations at the Institute.

Working group meeting on minor actinide containing targets and fuels, ITU Karlsruhe, 28-29 June 1994.

This was the fourth in the series of meetings of the Working Group on Targets and Fuels initiated following the Workshop on Partitioning and Transmutation of Minor Actinides held in the Institute in October 1989.
These meetings provide and international forum for the discussion of technical problems associated with all aspects of partitioning and transmutation. Presentations included national programmes, international co-operation and conceptual studies on different transmutation schemes, partitioning studies, post-irradiation examinations of minor actinides targets and fuels and different matrices containing minor actinides.
A major issue still to be resolved is that of determining the target nuclides to be transmuted. This compilation will influence the choice of partitioning techniques and transmutation schemes. It was clear that in the field fo partitioning one of the main goals remaining is the separation of actinides and lanthanides.

4. TERMINATION OF THE SHARED COST ACTION PROGRAMME OF THE EU ABOUT LMFBR SAFETY

The final results of the MOL7C /6 and /7 tests, which were organised as a Shared Cost Action (SCA) of the Reactor Safety Programme in the area of LMFBR safety, have been discussed in 1994 and documented in 2 final reports dated March 1995. This programme focused in particular on the behaviour of a LMFBR core under local flow coastdown accident conditions and consisted of 2 Actions, namely:

1- the in-pile study of the propagation potential of local subassembly faults, with emphasis on clad failure monitoring, damage evaluation and post-fault inherent coolability (Action 1 - Local subassembly faults / "MOL7C Programme") with the final Report "In-pile local blockage experiments MOL 7C/6 and 7"compiled by K. Schleisiek and J. Aberle (FZK, March 1995)
2 - the fission product inventory and the radiological impact of irradiated pin failures on the environment (Action 2 - Source term measurements / "Measurement of fission products in the experiments MOL7C/6 and 7") with the final Report "MOL 7C/6 and 7 / Measurement of fission products"compiled by A. Verwimp (SCK/CEN Mol, March 1995)

This SCA involved 2 senior partners, namely FZK and SCK/CEN Mol, JRC/STI Ispra being only a junior partner.

The general aim of the Mol 7C experiments in the BR2 reactor was to investigate the failure events and the inherent coolability conditions in a fuel bundle subjected to a local (partial) coolant flow blockage in the fissile zone. Operation of reactors with small number of failed fuel rods is desirable for economic reasons, but safety must never be prejudiced. Therefore the behaviour of failed fuel during accident is of primary interest and fuel simulation codes are essential in establishing an operational set of safety measures. The present MOL7C experiments fit clearly the current need for improved techniques which should help establish operational limits for fuel rods and develop higher performance materials.

To reach these objectives it is crucial :

1 - to better understand the kinetics of sodium-fuel chemical and thermomechanical interactions, having in mind the impact of contamination on reactor maintenance
2 - to examine the mechanism of delayed neutron (DN) precursor release so that shutdown limits can be established, that is:

- to demonstrate that after release of fission gases, fuel exposure to primary sodium can be readily detected by delayed neutron (DN) precursor release. Sufficient delayed neutron detector signals should be generated by any pin failure in the core, enabling one to detect in time any severe local loss-of-cooling event.

3 - to examine the potential for sweep-out of the released fuel as well as the damage propagation mechanism from pin-to-pin and from bundle-to-bundle
4 - to examine the behaviour of the fuel bundle during continued reactor operation with breached oxide fuel pins, that is:

- to examine how failed fuel pins and the neighbouring intact pins survive after a local subassembly accident, i. e. in particular how the various failure modes are affected by the overstress state due to high power-to-flow conditions
- to demonstrate that, in the case of severe fuel pin damage, the dispersive effects will dominate against melt-through effects: the accident will not propagate towards rapid core deterioration nor will the core loose significant quantities of fuel and fission products. Hence a complete core meltdown out of the primary containment can be really considered as very remote in probability.

Additional objectives of the MOL7C experiments, besides the general improvement of the in-pile instrumentation, consist of a deeper investigation (that is: a better numerical modelling) of sodium boiling and voiding, of cladding melting and relocation, of fuel melting and relocation both in voided and unvoided regions of the core.
The two experiments Mol 7C/6 and /7 were performed in a special sodium loop in BR2 on June 20-22, 1988 and June 12-21, 1989, respectively. They consisted of local blockage experiments involving prototypical bundles of mixed oxide fuel (UO_2-PuO_2) pins with grid spacers and an active porous blockage (highly enriched UO_2 spheres with chromium coating) of 40 mm height and located slightly above the region of maximum power. The MOL7C/6 test was a 37 rod bundle experiment with central active porous blockage and high burnup values (maximum 10 at % or 94200 MWd/ton M). The MOL7C/7 test was a 19 fresh rod bundle experiment with the active porous blockage at the periphery of the bundle.

It can be generally stated that the main objectives of the experiment were attained, that is:

1. All detection methods applied in the test recognized the fault in a prompt and distinct manner. In particular, the Delayed Neutron Detector systems are sufficient to initiate reliably automatic shutdown of the reactor. The MOL7C/6 test showed DND signal amplitudes which were even several orders of magnitude above the usual trip level of LMFBRs and it showed it already during the initial phase of the transient - which means that similar events in a reactor would trigger the shutdown system sufficiently in time before any fault propagation.
2. During the critical first minute, when boiling starts and before the DND signal, the fault will not escalate to such a degree that the coolability of the fuel assembly as a whole is endangered.
3. The secondary objective, i.e. the demonstration of inherent safety, was only partly attained in the MOL7C/6 experiment because the post-test irradiation could not last for a long period of time.

From the Gamma spectrometry investigations, fuel-to-sodium transfer coefficients were determined for noble gases, aerosols, light metals and metalloids. These coefficients are of course provisional until the complete PIE will allow determination of the actual amount of defect fuel to be considered.

The range of burnups investigated is only 50 % of the target burnup of LMFBRs. The extrapolation to real reactor conditions needs obviously extreme care and more experimental information from the PIE. Further tests are needed to examine various important parameters like: different O/M ratio fuels, burnups, new claddings (like the non-swelling ferritic/martensitic alloys), prototypic lengths, annular pin designs (with fabricated central void regions), midplane and multiple failures.

In a not too far future from now, it is hoped that better theoretical prediction tools will be available for both early subassembly desintegration and FP release to the environment, through a greater effort on code validation. Parameters such as the energy transfer from the fuel to the sodium and the subsequent mechanical work on the surrounding structures will then be calculated with the desired accuracies.

LIST OF PARTICIPANTS

Ainsworth, K.F.	British Nuclear Fuels plc, BNFL, Risley, Warrington, Cheshire WA3 6AS, United Kingdom
Bhoje, S.B. *(Chairman)*	Indira Gandhi Centre for Atomic Research, Kalpakkam 603 102, Tamil Nadu, India
Bramman, J.I.	"KIRKLEES", 30, Bruntcliffe Lane, Morely, LEEDS LS27 9LL, United Kingdom
Brogli, R.	Paul Scherrer Institute, PSI CH-5232 Villigen, Switzerland
Buksha, Yu.K.	Institute of Physics and Power Engineering, Bondarenko Sq. 1, 249020 Obninsk, Kaluga Region, Russian Federation
Cho, Mann	Korea Atomic Energy Research Institute (KAERI), P.O. Box 105, Yusong, Taejon, Republic of Korea
Heusener, G.	Forschungszentrum Karlsruhe Technik und Umwelt, Postfach 3640, D-76021 Karlsruhe, Germany
Itoh, K.	Reactor Development Project, Power Reactor and Nuclear Fuel Development Corporation, 1-9-13 Akasaka, Minato-ku, Tokyo 107, Japan
Kim, Young-Cheol.	Korea Atomic Energy Reserch Institute (KAERI), P.O. Box 105, Yusong, Taejon, Republic of Korea
Kupitz, J.	Nuclear Power Technology Development Section, Division of Nuclear Power, International Atomic Energy Agency, Wagramerstrasse 5, P.O. Box 100, A-1400 Vienna, Austria
Levitin, V.	Mangishlak Power Plant, MAEK, 466200 Aktau, Kazakhstan

Pedersen, T.	Nuclear Power Technology Development Section, Division of Nuclear Power, Wagramerstrasse - 5, P.O. Box 100 A-1400 Vienna, Austria
Picker, C.	AEA Technology, Risley, Warrington WA3 6AT, United Kingdom
Rigoleur, C.Le.	DER/CEA Cadarache, F-13108 Saint-Paul-lès-Durance - Cedex, France
Rinejski, A. *(Scientific Secretary)*	Nuclear Power Technology Development Section, Division of Nuclear Power, International Atomic Energy Agency, Wagramerstrasse 5, P.O. Box 100, A-1400 Vienna, Austria
Ryzhov, V.	Department of Nuclear Reactors, Ministry of Russian Federation for Atomic Energy, Staromonetny Pereulok 26, Moscow, Russian Federation
Semenov, B.A.	Department of Nuclear Energy and Safety, International Atomic Energy Agency, Wagramerstrasse 5, P.O. Box 100, A-1400 Vienna, Austria
Sukhanov, G.	Division of Nuclear Fuel Cycle and Waste Management, International Atomic Energy Agency, Wagramerstrasse 5, P.O. Box 100, A-1400 Vienna, Austria
Tavoni, R.	ENEA, Energy Department, Via Martiri di Monte Sole - 4, I-40129 Bologna, Italy
Xu, Mi	Department of Fast Reactor Engineering, China Atomic Energy Institute, P.O. Box 275 (95), 102413 Beijing, China